Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design
Docket No. 52-006

NUREG-1793
Supplement 2

Division of New Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This report supplements the final safety evaluation report (FSER) for the AP1000 standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1793 in September 2004, and NUREG-1793 Supplement 1 in December 2005, to document the NRC staff's technical review of the AP1000 design. The application for the AP1000 design was submitted on June 28, 2002, by Westinghouse Electric Corporation (Westinghouse) in accordance with Subpart B, "Standard Design Certifications," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, and Appendix 0, "Standardization of Design: Staff Review of Standard Designs." This supplement documents the NRC staff's review of Westinghouse's changes to the AP1000 design documentation in the design control document (DCD) since the issuance of Supplement 1 of the FSER. On the basis of the evaluation described in the AP1000 FSER (NUREG-1793, NUREG-1793 Supplement 1) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 19 to the AP1000 DCD) are acceptable and that Westinghouse's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

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1. INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

Supplement 2 to NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," addresses a revision to the AP1000 design control document (DCD) to reflect design changes submitted by Westinghouse Electric Company (the applicant) after the U.S. Nuclear Regulatory Commission (NRC) certified the design in Appendix D, "Design Certification Rule for the AP1000 Design," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, certifications, and approvals for nuclear power plants." The current review involves an amendment to the AP1000 design certification (DC), as documented in proposed changes to the AP1000 DCD through Revision 19.

Background

The certified AP1000 design, addressed in Appendix D to 10 CFR Part 52, has a nuclear steam supply system (NSSS) power rating of 3,415 megawatts thermal (MWt), with an electrical output of at least 1,000 megawatts electric (MWe). Prior to approval of the DC amendment, Revision 15 of the AP1000 DCD documented the approved design; NUREG-1793, issued September 2004, and Supplement 1, issued December 2005, documented the NRC staff's approval of this design.

From March 2006 through May 2007 (the preapplication period), NuStart and the applicant provided the NRC with technical reports (TRs) for preapplication review in an effort to: (1) close specific, generically applicable COL information items in the AP1000 certified standard design; (2) identify standard design changes resulting from the AP1000 detailed design efforts; and (3) provide specific standard design information in areas or for topics where the AP1000 DCD was focused on the design process and acceptance criteria. Appendix H, "Technical Reports," to this report includes a list of these TRs. The TRs include proposed revisions to the DCD and supporting information providing the basis for acceptability of the changes.

The application submitted on May 26, 2007, which transmitted Revision 16 to the DCD, was also supplemented by letters dated October 26, November 2, and December 12, 2007, and January 11 and January 14, 2008. The staff notified the applicant, in a letter dated January 18, 2008, that it accepted the May 26, 2007, application, as supplemented, for docketing. The January 18, 2008, letter included a *Federal Register* Notice (FRN) that provided public notification that the NRC had accepted the May 26, 2007, application, as supplemented, for docketing and that a future *Federal Register* Notice would provide an opportunity to comment on the proposed rulemaking.

In a letter dated September 22, 2008, the applicant submitted Revision 17 to the AP1000 DCD. The staff's review also included other design changes identified by the applicant following submittal of Revision 17, associated with Interim Staff Guidance (ISG) DC/COL-ISG-11, "Interim Staff Guidance Finalizing Licensing-basis Information," as detailed in Section 1.15, herein. On December 1, 2010, the applicant submitted Revision 18 to the DCD. Revision 18 incorporated the ISG design changes as well as the DCD changes to resolve confirmatory items from the Advanced Final Safety Evaluation (AFSE). Revision 19, submitted on June 13, 2011, includes additional DCD changes resulting from the staff's review of Revision 18. Revision 19 is a complete DCD and includes the relevant information from the certified design (Revision 15) that was not modified by subsequent revisions, and the net result of the changes from Revision 16 to 17 to 18 to 19.

Since this is a supplement to the previous safety evaluation report (SER), the staff's review of the application was based on the proposed changes included in Revisions 16 through 19 of the DCD. Material from Revision 15 and earlier is evaluated in the original NUREG-1793 or Supplement 1. This SER supplement is applicable to Revision 19 of the DCD, which is the revision intended for certification in the final rule. Individual SER sections may refer to specific revisions other than Revision 19 depending on the context; however, if information was added in Revisions 16, 17 or 18 and was not further modified, it is part of Revision 19.

1.1.1 Metrication

This report conforms to the Commission's policy statement on metrication published in the FR on June 19, 1996. Therefore, measures are expressed as metric units, followed by English units in parentheses. An example of a typical conversion would be as follows: The unit of air volume flow is measured in standard cubic meters per second (m^3/s) at 101 kilopascal (kPa) and 20 °Celsius (C) (standard cubic feet per minute (ft^3/min) at 14.7 pounds-force per square inch absolute (psia) and 68 °Fahrenheit (F).

1.1.2 Proprietary Information

This report references Westinghouse reports. Some of these reports and communications include information that the applicant requested be exempt from public disclosure, as provided by 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." For each such report, the applicant provided a nonproprietary version, similar in content except for the omission of the proprietary information. The staff based its findings on the proprietary versions of these documents, which are those primarily referenced throughout this report. Table 1.6-1 of Chapter 1, Tier 2, of the DCD lists all of the proprietary reports referenced that are viewed as part of the licensing basis for the AP1000 design.

Within certain chapters of this report, the staff needed to present proprietary information for completeness. In these chapters, the proprietary information was subsequently redacted in order to make this report publicly available but references are provided to the proprietary version of the chapter for those individuals permitted to review the proprietary information.

1.1.3 COL Applicants Referencing the AP1000 Design

Future applicants referencing the AP1000 standard design for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL) referencing the AP1000 design, the staff will evaluate, for each plant-specific application, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. COL applicants will also be subject to the requirements of 10 CFR Part 52, Subpart C, "Combined Licenses," and any requirements resulting from the staff's review of this standard design. Throughout the DCD, the applicant identified matters to be addressed by plant-specific applicants as "combined license information." This report generally refers to such matters as "COL action items" throughout (see also Section 1.9 below).

1.1.4 Additional Information

Parts of the DCD include summary tables (e.g., Introduction Table 1-1, Tables 1.6-1, 1.8-2, Appendix 1A of Chapter 1 of Tier 2) and drawings (e.g., figures in Section 1.2, Tier 2) that reflect proposed changes in the DCD to conform to changes in other chapters. Determinations about acceptability of those changes depend on conclusions to be documented in other chapters of the final safety evaluation report (FSER).

This FSER includes appendices to assist the reader. Appendix A provides a preapplication chronology of the principal actions, and submittals related to the processing of the AP1000 application; and Appendix B provides the post-application chronology. Appendix C of this report includes a list of references for the FSER; Appendix D lists the definitions of the acronyms and abbreviations; Appendix E lists the principal technical reviewers who evaluated the amendment to the AP1000 design; Appendix F provides an index of the applicant's technical reports (TRs); and Appendix G provides an index of the applicant's responses to requests for additional information (RAIs). Appendix H of this report includes a copy of the letters received from the Advisory Committee on Reactor Safeguards providing the results of its review of the safety evaluation chapters.

The NRC licensing project managers assigned to the AP1000 DC amendment review are Perry Buckberg and David H. Jaffe (Lead Project Managers), William Gleaves, Sikhindra Mitra, Phyllis Clark, Patrick Donnelly, Brian Anderson, and Terri Spicher. They may be reached by calling (301) 415-7000 or by writing to the U.S. Nuclear Regulatory Commission, Office of New Reactors, Washington, DC 20555-0001.

1.2 General Design Description

The DCD through Revision 19 includes a complete description of the AP1000.

1.3 Comparison with Similar Facility Designs

The AP1000 standard design includes many features that are not found in the designs of currently operating reactors. For example, a variety of engineering and operational improvements provides additional safety margins and addresses Commission policy statements regarding severe accidents, safety goals, and standardization. The most significant improvement to the design is the use of safety systems for accident prevention and mitigation that rely on passive means, such as gravity, natural circulation, condensation and evaporation, and stored energy. DCD Tier 2, Table 1.3-1, "AP1000 Plant Comparison with Similar Facilities," provides a detailed comparison of the principal design features of the AP1000 standard design with the certified AP600 design and a typical two-loop plant.

1.4 Summary of Principal Review Matters

The matters under review as part of the DC amendment process were mainly determined by the application. The DCD associated with the DC amendment identified changes, subject to review, by marginal lines. The remaining DCD text was from Revision 15 to the DCD and represented the unchanged elements of the DC of record referenced in Appendix D to 10 CFR Part 52. The staff did not repeat the review of the unchanged elements of Revision 15 to the DCD, in accordance with 10 CFR 52.63, "Finality of standard design certifications."

Examples of significant design changes include the following:

- extension of seismic spectra to soil conditions
- revisions to buildings for enhanced protection (such as for aircraft impact)
- protection system instrumentation update
- revisions to the electrical system (additional auxiliary transformer; change in direct current (dc) voltage)
- turbine manufacturer change
- sump screen design and analysis
- control room ventilation system
- increased assembly capacity in the spent fuel pool (SFP) (change in rack design)
- updated load handling systems
- additional waste-water monitor tanks
- integrated head package (IHP) revision
- revision to loss-of-coolant accident (LOCA) methods
- reactor internal changes (flow skirt addition)
- pressurizer shape change
- reactor coolant pump design
- addition of containment vacuum relief system
- completion of human factors engineering commitments
- revision to closure logic for component cooling system isolation
- reactor vessel structural support

The subjects in Supplement 2 to NUREG-1793 are organized in the same manner as NUREG-1793, which generally conforms to the organization of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." The only exception is Chapter 23, which documents the review of changes submitted late in the review process of design changes not prompted by NRC review activities. The absence in Supplement 2 to NUREG-1793 of a section that appeared in NUREG-1793 indicates that the staff did not repeat the review of this material as part of the DC amendment process because there were no DCD changes that affected its content.

1.5 Requests for Additional Information

RAIs are questions asked of the applicant by the staff concerning the application. The NRC sent the questions to the applicant by e-mail, and the applicant responded in letters to the NRC.

The nomenclature for RAIs concerning TRs took one of the following two forms:

- TRXX-YY, where XX was the TR number and YY was the RAI sequence number.

- TRXX-ABREV-YY, where ABREV was the abbreviation of the NRC review organization that initiated the question.

In early 2008, the staff began its review of the application using NUREG-0800. It then added the RAI designation RAI-SRPZ.ZZ-ABREV-YY, where Z.ZZ was the NUREG-0800 section number.

1.6 Open Items

In many cases, the applicant's responses to the RAIs resulted in the RAIs being closed in that the information that was provided was sufficient to resolve the issue. In those cases where the responses to the RAIs did not resolve the issue, the staff created an "Open Item [OI]" using the same conventions as used for RAIs with the prefix OI replacing the prefix RAI. The staff then issued a "Safety Evaluation with Open Items" for chapters of this report.

1.7 Confirmatory Items

Following issuance of the safety evaluation with open items, the applicant responded to the open items and all open items were resolved. Where information to resolve the open item would be in Revision 18 to the DCD (or a future activity by the applicant or the staff), the staff created a "Confirmatory Item" using the same conventions as used for open items with the prefix CI replacing the prefix OI. The staff then issued a safety evaluation with confirmatory items, also referred to as an AFSE for each chapter. Upon receipt of Revision 18 to the DCD, the staff confirmed that the information required to resolve the confirmatory items was in Revision 18 to the DCD or, where necessary, in Revision 19. The staff is issuing the final SER as Supplement 2 to NUREG-1793 which removes discussion about the resolved confirmatory items.

1.8 Index of Exemptions

There are no exemptions associated with the DC amendment; the exemptions that were part of the initial certification remain in effect.

1.9 COL Information Items

COL applicants and licensees referencing the certified AP1000 standard design must satisfy the requirements and commitments identified in the DCD. The AP1000 DCD identifies certain general commitments as "combined license information items." The COL information items are tabulated in Table 1.8-2 of the DCD, Tier 2. These COL information items relate to programs, procedures, and issues that are outside the scope of the certified design review. These COL information items do not establish requirements; rather, they identify an acceptable set of information to be included in a plant-specific safety analysis report. An applicant for a COL must address each of these information items in its application. An applicant may deviate from or omit these information items, provided that the deviation or omission is identified and justified in the plant-specific safety analysis report. As noted earlier, several of the DCD changes proposed in this amendment are for the purpose of responding, within the DCD, to COL information items from the original certification, so that no further action by a COL applicant would be necessary. In its evaluations, the staff may refer to these as COL action items, as was done in the original NUREG-1793. The DCD refers to these items as COL information items.

1.10 Technical Reports

The applicant submitted TRs for more than a year before providing the DC amendment application. The main purpose of the TRs was to provide the basis for proposed changes to the AP1000 DCD, and most TRs included marked-up DCD pages to show where these proposed changes would occur. TR-134, "AP1000 DCD Impacts to Support COLA Standardization," APP-GW-GLR-134, through Revision 5, followed the submittal of Revision 16 to the AP1000 DCD. The purpose of TR-134 was to show the cumulative changes to the DCD, following Revision 16, from all sources, including the submittal of and changes to TRs (and similar documents referred to as "impact reports") and responses to RAIs.

1.11 Criteria of 10 CFR Part 52, Section 52.63(a)(1)

In 2007, the Commission was involved in rulemaking in 10 CFR Part 52. The rulemaking included a new 10 CFR 52.63, which would provide criteria for a rulemaking to amend a DC. The rule in 10 CFR 52.63(a)(1) states in part:

> …the Commission may not modify, rescind, or impose new requirements on the certification information, whether on its own motion, or in response to a petition from any person, unless the Commission determines in a rulemaking that the change:
>
> (i) Is necessary either to bring the certification information or the referencing plants into compliance with the Commission's regulations applicable and in effect at the time the certification was issued;
>
> (ii) Is necessary to provide adequate protection of the public health and safety or the common defense and security;
>
> (iii) Reduces unnecessary regulatory burden and maintains protection to public health and safety and the common defense and security;
>
> (iv) Provides the detailed design information to be verified under those inspections, tests, analyses, and acceptance criteria (ITAAC) which are directed at certification information (i.e., design acceptance criteria);
>
> (v) Is necessary to correct material errors in the certification information;

(vi) Substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; or

(vii) Contributes to increased standardization of the certification information.

These criteria, items (i) through (vii) above, were adopted as part of the final rule for 10 CFR Part 52, on August 28, 2007.

In revising the DCD, the applicant proposed numerous changes to the AP1000 design, including, but not limited to, minor component design details, replacement of a design feature with another having similar performance (e.g., turbine manufacturer, power for the auxiliary boiler), and changes allowing additional capability for operational flexibility (e.g., liquid waste holdup tanks, unit reserve transformer). The applicant included in its application a detailed list of each DCD content change and the basis under 10 CFR 52.63(a)(1) that supported including that change in the amendment. The staff considered these bases and formed its own judgment on applicability of the criteria to the changes under review. More than one criterion may be satisfied for any particular change; it is only necessary that one criterion be met to support inclusion in the amendment in accordance with 10 CFR 52.63. In a few instances, the staff concluded that none of the criteria were met and thus rejected inclusion of those changes. For those changes remaining in the scope of the amendment, the NRC concluded that at least one of the criteria in 10 CFR 52.63(a) is met and therefore did not constitute a violation of the finality provisions in that section.

The proposed rule includes a list of the changes to the DCD that the staff considers to be the most significant, the location in this SER where the change is evaluated and the principal criterion in 10 CFR 52.63(a)(1) that was cited as the basis for the change. Due to the significance of these particular changes, the NRC addressed the criteria of 10 CFR 52.63(a)(1) in detail in the proposed rule notice. Most of these changes cited criterion (vii) “contributes to increased standardization of the certification information.” The NRC further stated that increased standardization is realized through changes that are included in the amendment and incorporated by reference and, therefore, do not need to be handled as departures by each of the COL applicants. Other changes that provide more detailed information within the DCD (as for instance where a COL information item was resolved) also contribute to increased standardization.

For other changes evaluated in the SER, the finality criteria are addressed in varying degrees of detail in the notice and in this report.

1.12 DCD Editorial Changes and Changes for Consistency

The applicant has proposed numerous changes to the DCD that can be categorized as editorial changes or changes for consistency as follows:

- Editorial changes correct a spelling, punctuation, or similar error and result in text that has the same essential meaning; these changes are not subject to a safety evaluation.

- Changes for consistency must be made to the text in one or more instances to achieve uniformity. These changes require a safety evaluation, which is located in the SER where the subject is normally addressed via NUREG-0800 (e.g., a change to the type of reactor coolant pump motor is evaluated in Chapter 5 of this report; however, for consistency, a change to the description of the motor is needed elsewhere in the DCD, where the type of motor is described).

The revision change roadmap in the front of Revisions 16 through 19 shows the specific pages in the DCD where such changes were made.

Editorial changes to the DCD do not require a safety evaluation because they do not result in a change to any regulatory requirement. In accordance with 10 CFR 52.63(a)(1)(vii), these proposed changes are acceptable, since they contribute to standardization by making these changes on an individual basis unnecessary for subsequent COL applicants. Changes that generated additional changes that were needed for consistency are acceptable for reasons described in this safety evaluation in sections where these subject matters are normally addressed via NUREG-0800. Internal consistency in the DCD is needed so that it is an accurate document, and thus the conforming changes are acceptable.

1.13 Editorial Format Changes Related to COL Applicant and COL Information Items

In a letter dated June 6, 2007, the applicant submitted TR-130, "Editorial Format Changes Related to Combined License Applicant and Combined License Information Items," APP-GW-GLR-130, Revision 0. The revision change roadmap located in the front of Revision 16 shows the specific pages in the DCD where such changes were made. TR-130 proposed two classes of changes to the DCD:

- Editorial Format Changes Related to Combined License Applicant. In sections of the DCD that refer to a COL applicant's or COL holder's commitments (other than "Combined License Information" sections), the reference to a COL applicant or COL holder is deleted and replaced by a reference to the DCD section where the commitment is discussed. Certain sections in DCD Chapters 2 and 14 have not been changed, in this regard, as described in TR-130. The staff has reviewed these proposed DCD changes described in TR-130 and concludes that no changes to COL applicant or COL holder commitments result from the proposed changes, since the statement of the COL information items remains unchanged. Since the proposed changes add useful information, by referencing the DCD section that discusses the commitments, the overall result is an improvement in the usability of the DCD.

- Editorial Format Changes Related to Combined License Information Items. It has been the applicant's practice, when closing COL information items, to simply note that the item is "completed" when the commitment has been satisfied. In TR-130, the applicant has proposed adding information to the statement of the COL information items indicating how the commitment was completed (e.g., by identifying a Westinghouse document) and what tasks, if any, remain to be accomplished by the COL applicant or holder. Similar information would also be added to DCD, Tier 2, Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items." The staff has reviewed these proposed DCD changes described in TR-130 and concludes that no changes to COL applicant or COL holder commitments result from the proposed changes. Useful information is added to show how commitments were satisfied and what, if anything, is still needed to satisfy the remaining commitments. Since the proposed changes add useful information, the overall result is an improvement in the usability of the DCD.

In accordance with 10 CFR 52.63(a)(1)(vii), these proposed changes are acceptable, since they contributed to standardization by making these changes unnecessary for subsequent COL applicants.

1.14 Severe Accident Mitigation Design Alternatives

In 10 CFR 51.55(b), "Environmental report—construction permit, early site permit, or combined license stage," the NRC requires each applicant for an amendment to a DC to submit a separate document entitled, "Applicant's Supplemental Environmental Report—Amendment to Standard Design Certification." The environmental report must address whether the design change that is the subject of the proposed amendment either causes a severe accident mitigation design alternative (SAMDA) previously rejected in an environmental assessment to become cost-beneficial, or results in the identification of new SAMDAs that may be reasonably incorporated into the DC. In a letter dated September 21, 2007, the applicant submitted TR-135, "AP1000 Design Change Proposal Review for PRA and Severe Accident Impact," APP-PRA-GER-001, Revision 0. In TR-135, the applicant documented the review of all design-change proposals approved since the DC and evaluated their potential impact on the AP1000 probabilistic risk assessment (PRA). The staff has reviewed TR-135 and supplemental letters dated October 26 and November 9, 2010, and concludes that these design changes have no significant impact on the results of the AP1000 PRA. Chapter 19 presents the staff's review of changes to the PRA. Consequently, the AP1000 SAMDA analyses remain valid: none of the previously evaluated SAMDAs is cost-beneficial. No new SAMDAs have been identified.

Based upon the above, the staff concludes that the applicant has complied with the requirements of 10 CFR 51.55(b) with regard to the application to amend the DC for the AP1000.

1.15 Changes to Regulatory Guides and Criteria

The applicant has submitted the following two TRs that, together, describe changes in the AP1000 DCD related to conformance to regulatory guides (RGs), Three Mile Island (TMI) issues, unresolved safety issues and generic safety issues, and advanced light-water reactor (LWR) certification issues since Revision 15:

- TR-129, "Changes to Conformance with Regulatory Guidance and Criteria," APP-GW-GLN-129, issued June 2007

- TR-141, "Regulatory Guide Conformance Changes," APP-GW-GLN-141, issued October 2007

Conformance to RGs, TMI issues, unresolved safety issues and generic safety issues, and advanced LWR certification issues are addressed in DCD, Tier 2, Sections 1.9.1 (and Appendix 1A), 1.9.3, 1.9.4 and 1.9.5, respectively.

TR-129 also proposes to add COL Information Item 1.9-1 to DCD, Tier 2, Table 1.8-2, “Summary of AP1000 Standard Plant Combined License Information Items,” and a new DCD, Tier 2, Section 1.9.1.5, “Combined License Information,” as follows:

> The Combined License applicant will address conformance with regulatory guides that are not applicable to the certified design or not addressed by the activities required by COL information items.

The list of RGs proposed by the applicant, as shown in Table 1.15-1, is the subject of proposed COL Information Item 1.9-1. COL applicants may supplement the list of RGs in Table 1.15-1 as needed. In addition, as part of an RAI, the staff may request COL applicants to address one or more additional RGs; otherwise, the staff finds the proposed COL information item to be acceptable, in accordance with 10 CFR 52.63(a)(1)(vii), in that it contributes to standardization by making it unnecessary for individual COL applicants to request the associated changes.

DCD, Tier 2, Table 1.9-1, “Regulatory Guides/DCD Section Cross-References,” and Appendix 1A discuss details regarding conformance to RGs, including the changes proposed by TR-129 and TR-141 and as a result of other design changes. NUREG-1793, Chapter 1, did not present an evaluation of the applicant’s conformance to RGs with regard to the AP1000 and, similarly, no evaluation is presented herein regarding changes to these positions in this tabular form. Conformance to RGs is evaluated in the specific sections of the SER where the DCD material concerning the RG is discussed. For example, RG 1.82, Revision 3, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident” is discussed in DCD Section 6.2.2 and evaluated in Section 6.2.1.8 of this report.

Table 1.15-2 includes a list of changes to regulatory criteria (TMI issues, unresolved safety issues and generic safety issues, and advanced LWR certification issues) where the changes proposed in TR-129 and TR-141 are editorial, are required for consistency with proposed changes elsewhere in the DCD, or provide additional useful information. These proposed changes have no impact on safety-related structures, systems, components (SSCs), or other design aspects and are acceptable, in accordance with 10 CFR 52.63(a)(1)(vii), in that they contribute to standardization by making it unnecessary for individual COL applicants to request the associated changes.

Finally, Table 1.15-3 includes changes to regulatory criteria that are addressed elsewhere in this SER and indicates the location in this report. Also, the location of the staff’s evaluation as documented in the SER is indicated in Table 1.15-3.

1.16 Design Changes Proposed in Accordance with Interim Staff Guidance (ISG)-11

DC/COL-ISG-11 describes the staff position regarding the control of licensing-basis information during and following the initial review of applications for DCs. It describes the categories of design changes that applicants should not defer until after the issuance of the DC rule. These criteria are presented in Chapter 23 of this report.

Chapter 23 addresses new design changes, proposed in accordance with DC/COL-ISG-11 that were then included in Revision 18 of the AP1000 DCD. The design changes that are evaluated in Chapter 23 do not constitute all of the changes that the applicant included in DCD, Revision 18. Rather, the design changes evaluated in Chapter 23 are in addition to those that the applicant has submitted to the NRC as a part of responses to RAIs or SER open items.

Organizationally, Chapter 23 is different from other SER chapters in that these design changes consider all aspects of a design together (i.e., electrical, instrumentation and control (I&C), piping, etc.) in one section rather than including various aspects of a design in separate chapters. Those who use this SER should also refer to Chapter 23 in that the analyses included therein supplement the analyses found elsewhere in this report.

1.17 Tier 2* Information

Information designated as Tier 2* (Tier 2 Information Requiring NRC Approval for Change) is identified in the DCD by brackets, italics, and a footnote noting that prior NRC approval is needed for any departure from that information. It is also summarized in Table 1-1 of the DCD. The rule text in Appendix D to 10 CFR Part 52 lists the topic areas with Tier 2* information. During the review of the amendment request, some changes to the material designated as Tier 2* occurred, as summarized below.

In DCD Chapter 3, “Design of Structures, Components, Equipment and Systems,” Sections 3.8 and Appendix 3H, as originally certified, considerable information about critical sections of the structures was designated as Tier 2*. This included load combinations, specific analytical results (loads and moments), and resultant structural reinforcement thicknesses. The staff determined that having Tier 2* designation on analytical results (with several significant digits) was unduly restrictive. As a result, the DCD tables with Tier 2* information were revised to retain the designation on loads and reinforcements (with some tolerance), but removed the results from the scope of Tier 2*. The rule text did not change for this reason.

In addition, the staff determined that other structural information about aspects of the design, such as the shield building and containment penetrations, should be designated as Tier 2*. Multiple locations in Section 3.8 and Appendix 3H are now so marked (and listed in Table 1-1 of the DCD Introduction). In addition, a referenced technical report (GLR-602) that includes proprietary information about the shield building also has Tier 2* information (see Table 1.6-1 of the DCD). Conforming changes to the final rule language will be made as needed to include the type of information in the sections of the rule that identify Tier 2* information. The Tier 2* designation for these structural details would expire at first full power.

The staff requested that the applicant add Tier 2* designation to the specification of the reactor coolant pump (RCP) characteristics, a new Tier 2* item that does not expire. This information appears in DCD Chapter 5, “Reactor Coolant System and Connected Systems,” and it was added to Table 1-1 of the Introduction. A new item was included in the proposed rule to reflect this change.

In Revision 15 of the AP1000 DCD (Sections 3.8.2.2 and 5.2.1.1), the specific Edition and Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III were designated as Tier 2* information. At the time of the initial DC, the staff accepted the 1998 Edition up to and including the 2000 Addenda of the ASME Code, Section III (except for piping design, which uses the 1989 Edition including the 1989 Addenda) as Tier 2* to ensure that the ASME Code, Section III piping seismic design rules that the staff did not fully accept would not be used for completing the AP1000 piping design without first obtaining NRC approval. The NRC issued a final rule amending 10 CFR 50.55a, “Codes and standards,” (64 FR 51370 dated September 22, 1999) that included a condition in 10 CFR 50.55a(b)(1)(III), “Seismic design of piping,” prohibiting the use of these piping seismic design rules that first appeared in the 1994 Addenda of the ASME Code, Section III. This limitation remained in effect and applicable up to and including the 2004 Edition (referenced in 10 CFR 50.55a). As a result of the NRC establishing the limitation in 10 CFR 50.55a(b)(1)(III) prohibiting those portions of the ASME Code, Section III related to revised seismic design rules, the need to designate the specific Edition and Addenda of the ASME Code, Section III as Tier 2* became redundant and unnecessary. However, the NRC is requiring that certain DCD provisions, related to piping design that was already marked as Tier 2*, remain with that designation.

For design of components as discussed in DCD Section 5.2.1.1, the staff concluded that the Tier 2* designation was not necessary for the specific ASME Code Edition and Addenda, as listed in Item VIII.B.6.c (2) of Appendix D to 10 CFR Part 52. Subsequent to the certification, 10 CFR 50.55a was modified to include provisions in paragraphs (c)(3), (d)(2) and (e)(2), for reactor coolant pressure boundary (RCPB), Quality Group B components, and Quality Group C components, respectively. These paragraphs provide the controls on use of later Edition/Addenda to the ASME Code, Section III through the conditions NRC established on use of paragraph NCA-1140 of the ASME Code. As a result, these rule requirements would adequately control the ability of a licensee to use a later Edition of the ASME Code and Addenda, such that the Tier 2* designation is not necessary for components. Thus, the item in VIII.B.6.c (2) for the ASME Code was proposed to be modified in the proposed rule to be more limited in scope. In addition, Item VIII.B.6.c (2) now also refers to ASME Code cases; Table 5.2-3 of the DCD lists the applicable Code cases and which ones are Tier 2*.

The NRC is retaining the Tier 2* designation for the ASME Code Edition applicable to containment design in VIII.B.6.c (14). The designation of the Edition and Addenda of the ASME Code, Section III, for completing the construction of the AP1000 steel containment is Tier 2.

The ACRS review highlighted the significance of certain assumptions about debris in containment to the adequacy of long-term core cooling, and a concern that the values not be revised with substantial additional testing and analysis. As a means of emphasizing this, the applicant proposed to designate the key information as Tier 2*, to require prior NRC approval, in a letter dated February 23, 2011. This change is included in Revision 19. The NRC agrees that this is a prudent change and will modify the final rule language to reflect this addition, as a Tier 2* item without expiration at fuel load.

The staff requested that the applicant revise the Tier 2* expiration for human factors engineering in DCD Chapter 18, “Human Factors Engineering,” from no expiration to expiration at initial power operation. The rule item thus was proposed to be moved from paragraph VIII.B.6(b) to VIII.B.6(c) in the proposed rule.

The changes in Tier 2* information described above have been incorporated in Revision 19 to the DCD.

Table 1.15-1. Regulatory Guides to be Addressed by COL Applicants

- RG 1.86, "Termination of Operating Licenses for Nuclear Reactors," Revision 0
- RG 1.111, "Methods for Estimating Atmosphere Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1
- RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," Revision 1
- RG 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors," Revision 0
- RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2
- RG 1.162, "Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels," Revision 0
- RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 0
- RG 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," Revision 0
- RG 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," Revision 0
- RG 1.184, "Decommissioning of Nuclear Power Reactors," Revision 0
- RG 1.185, "Standard Format and Content for Post-shutdown Decommissioning Activities Report," Revision 0
- RG 1.186, "Guidance and Examples of Identifying 10 CFR 50.2 Design Bases," Revision 0
- RG 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," Revision 0
- RG 5.9, "Specifications for Ge (Li) Spectroscopy Systems for Material Protection Measurements Part 1: Data Acquisition Systems," Revision 2

Table 1.15-2. Changes to Regulatory Criteria (Changes are Editorial, Required, or Provide Additional Useful Information)

Item	Issue	Acceptability
1	Revise Footnote f. to Table 1.9-2	Editorial format changes related to Combined License applicant
2	Revise the response to 1.9.3, (2)(i), Simulator Capability (NUREG-0933, Item I.A.4.2)	Same as Item 1
3	Revise the response to 1.9.3, (2)(ii), Plant Procedures (NUREG-0933, Item I.C.9)	Same as Item 1
4	Revise the response to 1.9.3, (2)(xxv), Emergency Response Facilities (NUREG-0737, Item III.A.1.2)	Same as Item 1
5	Revise the response to 1.9.3, (3)(vii), Management Plan (NUREG-0933, Item II.J.3.1)	Same as Item 1
6	Revise the response to 1.9.4.2.3, II.K.1(10), Review and Modify Procedures for Removing Safety-related Systems from Service	Same as Item 1
7	Revise the final paragraph of the response to A-31, Residual Heat Removal Requirements	Same as Item 1
8	Revise the response to 1.9.4.2.3, Issue 79, Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Same as Item 1
9	Revise the final paragraph of the response to 1.9.4.2.3, Issue 113, Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers	Same as Item 1
10	Revise the ninth bullet under Task 3 of the response to 1.9.4.2.3, Issue 135, Integrated Steam Generator Issues	Same as Item 1
11	Revise the sixth bullet of the response to 1.9.5.1.5, Station Blackout	Same as Item 1
12	Revise the response to 1.9.5.1.15, In-Service Testing of Pumps and Valves	Same as Item 1
13	Revise the response to 1.9.5.2.6, Tornado Design Basis	Same as Item 1
14	Revise the response to 1.9.5.3.7, Simplification of Off-Site Emergency Planning	Same as Item 1
15	Revise Section 1.9.6, References	Same as Item 1

Table 1.15-3. Changes to Regulatory Criteria (Addressed Elsewhere in this SER)

Items	Issues	Addressed in SER
1	Revise reference to QME testing standard in Issue 87	Section 3.9.6
2	Revise the response to 1.9.4.2.3, Issue 103, Design for Probable Maximum Precipitation	Sections 2.4.3 and 2.4.4
3	Revise 1.9.4.2.3, Issue 191, Assessment of Debris Accumulation on PWR Sump Performance	Section 6.2.1.8
4	Revise 1.9.4.2.4, HF4.4, Guidelines for Upgrading Other Procedures	Section 13.5
5	Revise the ninth bullet of the response to 1.9.5.1.5, Station Blackout	Section 8.3.1.2
6	Revise the response to 1.9.5.2.14, Site-Specific Probabilistic Risk Assessments (PRAs)	Section 19.1.5

2. SITE ENVELOPE

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Introduction

AP1000 design control document (DCD) Section 2.2.1 states that the combined license (COL) applicants referencing the AP1000 certified design will provide site-specific information related to the identification of hazards within the site vicinity, including an evaluation of potential accidents due to nearby industrial, transportation, and military facilities.

2.2.2 Evaluation

The U.S. Nuclear Regulatory Commission (NRC) staff has prepared safety evaluation report (SER) Section 2.2 in accordance with the review procedures described in the March 2007 revision of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity," and Section 2.2.3, "Evaluation of Potential Accidents," using information presented in the revised AP1000 DCD. Since the AP1000 design specific standard chemicals were not evaluated for explosion hazard, the staff has requested in request for additional information (RAI)-SRP2.2-RSAC-01, that the applicant provide required information pertaining to hazards of explosive chemicals stored onsite. The applicant responded with proposed changes to the AP1000 DCD. The staff has reviewed the applicant's response and the proposed changes to the AP1000 DCD.

2.2.3 Description of Proposed Change

The applicant identified the proposed changes to DCD Section 2.2 based on RAI-SRP2.2-RSAC-01, Revision 1. These changes included the description and evaluation of the AP1000 certified design-specific (standard) chemicals stored onsite for the explosion hazard. The applicant presented, for each explosive chemical, the minimum safe distance from the nearest structures, systems, and components (SSCs) that would not result in an overpressure in excess of 6.9 kilopascals (kPa) (1 pounds per square inch (psi)) from potential explosions and flammable vapor clouds (delayed ignition). The list of chemicals along with calculated minimum safe distances are presented in the proposed AP1000 DCD Table 2.2-1.

2.2.4 Applicable Regulations and Associated Acceptance Criteria

The relevant requirements of the NRC's regulations for these areas of review, and the associated acceptance criteria, are given in Sections 2.2.1, 2.2.2, and 2.2.3 of NUREG-0800, and are summarized below. Review interfaces with other NUREG-0800 sections can be found in Sections 2.2.1, 2.2.2, and 2.2.3.

1. Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(1), "Contents of applications; technical information," which requires a design certification (DC) applicant to provide site parameters postulated for the design. However, DC applications do not provide site characteristics because this information is site-specific and is not standard design-specific and, therefore, is addressed by the COL applicant. There are no

postulated site parameters for a DC related to Sections 2.2.1, 2.2.2, and 2.2.3 of NUREG-0800.

2. This regulatory basis is provided for information only since it applies to a COL applicant's final safety analysis report (FSAR) Sections 2.2.1 and 2.2.2. 10 CFR 100.20(b), "Factors to be considered when evaluating sites," which requires that the nature and proximity of man-made hazards (e.g., airports, dams, transportation routes, military and chemical facilities) be evaluated to establish site parameters for use in determining whether plant design can accommodate commonly occurring hazards, and whether the risk of other hazards is very low (applies to DCD Sections 2.2.1 and 2.2.2 only).

3. This regulatory basis is provided for information only since it applies to a COL applicant's FSAR Sections 2.2.1, 2.2.2, and 2.2.3. 10 CFR 52.79(a)(1)(iv), Contents of applications; technical information in final safety analysis report," as it relates to the factors to be considered in the evaluation of sites, which require the location and description of industrial, military, or transportation facilities and routes, and of 10 CFR 52.79(a)(1)(vi), as it relates to the compliance with 10 CFR Part 100, "Reactor site criteria," (applies to DCD Sections 2.2.1, 2.2.2 and 2.2.3).

Acceptance criteria are provided in NUREG-0800 to meet the above requirements:

1. This acceptance criterion for Section 2.2.1-2.2.2 of NUREG-0800 is provided for information only since it applies to a COL applicant's FSAR Section 2.2.1. Data in the safety analysis report (SAR) should adequately describe the locations and distances from the plant of nearby industrial, military, and transportation facilities and that such data are in agreement with data obtained from other sources, when available (applies to DCD Section 2.2.1 only).

2. This acceptance criterion for Section 2.2.1-2.2.2 of NUREG-0800 is provided for information only since it applies to a COL applicant's FSAR Section 2.2.2. Descriptions of the nature and extent of activities conducted at the site and in its vicinity, including the products and materials likely to be processed, stored, used, or transported, are adequate to permit identification of the possible hazards cited in Section III of Sections 2.2.1 and 2.2.2 of NUREG-0800 (applies to DCD Section 2.2.2 only).

3. This acceptance criterion for Section 2.2.1-2.2.2 of NUREG-0800 is provided for information only since it applies to a COL applicant's FSAR Section 2.2.2. Sufficient statistical data with respect to hazardous materials are provided to establish a basis for evaluating the potential hazards to the plant or plants considered at the site (applies to DCD Section 2.2.2 only).

4. This acceptance criterion for Section 2.2.3 of NUREG-0800 is provided for information only since it applies to a COL applicant's FSAR Section 2.2.3. Event Probability: The identification of design basis events (DBEs) resulting from the presence of hazardous materials or activities in the vicinity of the plant or plants of specified type is acceptable if all postulated types of accidents are included for which the expected rate of occurrence of potential exposures resulting in radiological dose in excess of the 10 CFR 50.34(a)(1), "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors," limits as it relates to the requirements of 10 CFR Part 100 is estimated to exceed the staff's objective of an order of magnitude of 10^{-7} per year (applies to DCD Section 2.2.3 only).

5. This acceptance criterion for Section 2.2.3 of NUREG-0800 is provided for information only since it applies to a COL applicant's FSAR Section 2.2.3. DBEs: The effects of DBEs have been adequately considered, in accordance with 10 CFR 100.20(b), if analyses of the effects of those accidents on the safety-related features of the plant or plants of specified type have been performed and measures have been taken (e.g., hardening, fire protection) to mitigate the consequences of such events (applies to DCD Section 2.2.3 only).

2.2.5 Evaluation

The staff reviewed the applicant's response to RAI-SRP2.2-RSAC-01, which included the proposed revision to AP1000 DCD, Section 2.2 pertaining to the description and evaluation of potential explosion hazards of explosive standard AP1000 design-specific chemicals stored onsite. The applicant evaluated the accidents involving potential explosions from the explosive chemicals stored onsite. Minimum safe distance not to exceed 1 psi peak incident overpressure to nearest critical plant structure is determined and presented in Table 2.2-1. The applicant concluded in this section that peak incident overpressure of 1 psi is not exceeded at the nearest SSC. The staff performed independent confirmatory analyses with conservative assumptions and using regulatory guide (RG) 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Revision 1 methodology and found that the results are comparable to those determined by the applicant. Therefore, the staff concludes that the applicant's methodology is reasonable, and the results and conclusions are acceptable. In a subsequent revision to the AP1000 DCD, the applicant included these changes in the DCD text.

2.2.6 Conclusions

The staff reviewed the applicant's response to RAI-SRP2.2-RSAC-01 and proposed revision to AP1000 DCD, Section 2.2. The results of the staff's technical evaluation of the information related to the evaluation of potential explosion hazard of explosive chemicals stored onsite are comparable to the results presented by the applicant in the revised AP1000 DCD. Therefore, the staff concludes that the applicant's analyses and conclusions are acceptable.

As set forth above, the applicant has identified potential explosion hazards of standard AP1000 design-specific chemicals stored onsite, and has appropriately determined those that should be considered in DBEs, and has demonstrated that the AP1000 design is adequately protected against potential design-basis events resulting from explosive chemicals stored onsite. The staff has reviewed the proposed information that included in the AP1000 DCD and, for the reasons specified above, concludes that the applicant has established that the AP1000 design meets the requirements of 10 CFR 52.47(a)(1) and also complies with 10 CFR 52.79(a)(1)(iv).

2.3 Meteorology

2.3.1 Regional Climatology

The revised AP1000 DCD changed some of the air temperature site parameters listed in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1. Table 2.3.1-1 of this SER presents these changes. Revision 17 changes are benchmarked against Revision 15, because Revision 15 is the version of the AP1000 DCD previously approved by the staff.

Table 2.3.1-1 Revisions to Air Temperature Site Parameter Values

TIER LEVEL	SITE PARAMETER	DCD REVISION 15	DCD REVISION 17
Tiers 1 and 2	maximum safety dry bulb with coincident wet bulb	115 °Fahrenheit (F)/80 °F (46.1 °Celsius (C)/26.7 °C)	115 °F/86.1 °F (46.1 °C/30.1 °C)
	maximum safety wet bulb (noncoincident)	81 °F (27.2 °C)	86.1 °F (30.1 °C)
Tier 2	maximum normal dry bulb with coincident wet bulb	100 °F/77 °F (37.8 °C/25.0 °C)	101 °F/80.1 °F (38.3 °C/26.7 °C)
	maximum normal wet bulb (noncoincident)	80 °F (26.7 °C)	80.1 °F (26.7 °C)

There were no changes in: (1) the minimum safety air temperature site parameter value (-40 °C (-40 °F)) presented in DCD Tier 1, Table 5.0-1; and (2) the minimum normal air temperature site parameter value (-23.3 °C (-10 °F)) presented in both DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1.

Revision 17 also made the following changes to the footnotes in DCD Tier 2, Table 2-1:

- Footnote (b) was expanded to clarify that: (1) the maximum normal values are 1-percent seasonal exceedance temperatures (June through September in the northern hemisphere), which are approximately equivalent to the annual 0.4-percent exceedance temperatures; and (2) the minimum normal value is the 99-percent seasonal exceedance temperature (December through February in the northern hemisphere), which is approximately equivalent to the annual 99.6-percent exceedance temperature.

- Footnote (g) was added to state that the containment pressure response analysis is based on a conservative set of dry-bulb and wet-bulb temperatures that envelop any conditions where the dry-bulb temperature is 46.1 °C (115 °F) or less and the wet-bulb temperature is less than or equal to 30.1 °C (86.1 °F).

These revisions relied on the following source documents:

- APP-GW-GLN-108, "AP1000 Site Interface Temperature Limits," Revision 2, September 2007

- APP-GW-GLE-036, "Impact of a Revision to the Current Wet Bulb Temperature Identified in Table 5.0-1 (Tier 1), and Table 2-1 (Sheet 1 of 3) of the DCD (Revision 16)," Revision 0, June 27, 2008

2.3.1.1 Evaluation

The staff has prepared SER Section 2.3.1 in accordance with the review procedures described in NUREG-0800 Section 2.3.1, using information presented in DCD Revision 17, APP-GW-GLN-108, APP-GW-GLE-036, and the applicant's responses to RAIs on APP-GW-GLN-108 and APP-GW-GLE-036. Where appropriate, the applicant has incorporated its RAI responses in Revision 17 of the DCD. Since the staff has reviewed the DCD Revision 17 and DCD Revision 17 includes the incorporation of the RAI responses, the staff considers the RAIs related to the DCD to be closed.

2.3.1.1.1 General Description

10 CFR 52.47(a)(1) requires in part that the standard DC application include the site parameters postulated for the design, and 10 CFR 52.79(d)(2) requires a COL application (FSAR) referencing a standard design to demonstrate that the site characteristics fall within the site parameters specified in the DC. AP1000 DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, present the list of AP1000 site parameters. If the FSAR does not demonstrate that the site characteristics fall within the site parameters specified in the DC, the COL application must include a request for an exemption or departure, as appropriate, that complies with the requirements of the referenced DC rule and 10 CFR 52.93, "Exemptions and variances."

SER Section 2.3.1 addresses the climatic site parameters (i.e., air temperature, wind speed, precipitation (snow and ice)) used as design bases for the AP1000. The list of Tier 1 site parameters includes maximum and minimum safety air temperature values, which are based on historical data and exceed peaks of less than 2 hours; the list of Tier 2 site parameters includes the same maximum and minimum safety air temperature values as well as maximum and minimum normal air temperature values, which are 1-percent seasonal exceedance values.

2.3.1.1.2 Description of Proposed Change

SER Table 2.3.1-1 lists the changes in air temperature site parameter values from DCD Revision 15 to DCD Revision 17. SER Table 2.3.1-1 shows that all the revised air temperature site parameter values are greater than before: the maximum safety coincident wet bulb increased 3.4 °C (6.1 °F) (from 26.7 °C (80 °F) to 30.1 °C (86.1 °F)), the maximum safety noncoincident wet bulb increased 2.8 °C (5.1 °F) (from 27.2 °C (81 °F) to 30.1 °C (86.1 °F)), the maximum normal dry bulb increased 0.5 °C (1 °F) (from 37.8 °C (100 °F) to 38.3 °C (101 °F)), the maximum normal coincident wet bulb increased 1.7 °C (3.1 °F) (from 25.0 °C (77 °F) to 26.7 °C (80.1 °F)), and the maximum normal noncoincident wet bulb increased 0.05 °C (0.1 °F) (from 26.7 °C (80 °F) to 26.7 °C (80.1 °F)).

The applicant used APP-GW-GLN-108 as its source document for the DCD Revision 16 changes in maximum safety noncoincident wet bulb (from 27.2 °C (81 °F) to 29.7 °C (85.5 °F)), maximum normal coincident wet bulb (from 25.0 °C (77 °F) to 26.7 °C (80.1 °F)), and maximum normal noncoincident wet bulb (from 26.7 °C (80 °F) to 26.7 °C (80.1 °F)). This document states that these modifications to air temperature site parameters better accommodate a broader range of conditions to encompass the potential sites for AP1000 plants. It also provides details on the effects of these changes to air temperature site parameters on a number of SSCs, such as the passive containment cooling system, the normal residual heat removal system, the spent fuel pool cooling system, the service water system, the component cooling water system, and the central chilled water system.

The applicant used APP-GW-GLE-036 as its source document for the subsequent changes in maximum safety coincident wet bulb (from 26.7 °C (80 °F) to 30.1 °C (86.1 °F)), maximum safety noncoincident wet bulb (from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F)), and maximum normal dry bulb (from 37.8 °C (100 °F) to 38.3 °C (101 °F)). This document states that these changes encompass more sites in the eastern United States, such as Levy County and Turkey Point. It also provides details on the effects of these changes to air temperature site parameters on the SSCs listed above.

2.3.1.1.3 Applicable Regulations and Associated Acceptance Criteria

Acceptance criteria regarding regional climatology site parameters, such as air temperature, are based on meeting the relevant requirements of General Design Criterion (GDC) 2, “Design Bases for Protection Against Natural Phenomena,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic licensing of production and utilization facilities.” GDC 2 states, in part, that SSCs important to safety must be designed to withstand the effects of natural phenomena without losing the ability to perform their safety functions.

GDC 2 also states that the design bases for these SSCs shall reflect, in part, appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

NUREG-0800 Section 2.3.1 states that the DC application should include ambient temperature and humidity statistics for use in establishing heat loads for the design of normal plant heat sink systems; post-accident containment heat removal systems; and plant heating, ventilation, and air conditioning systems. NUREG-0800 Section 2.3.1 also states that the climatic conditions identified as site parameters for DC applications should be representative of a reasonable number of sites that may be considered within a COL application and that a basis should be provided for each of the site parameters.

2.3.1.1.4 Evaluation

This SER section is limited to reviewing the appropriateness of the values chosen as air temperature site parameters; other SER sections (e.g., 5.4.7, 6.2.2, 9.1.3, 9.2.1, 9.2.2, and 9.2.7) review the effects of these changes to air temperature site parameters on SSCs.

To determine if the applicant’s revised air temperature site parameters are representative of a reasonable number of potential COL sites, the staff reviewed dry-bulb and wet-bulb data from the Weather Data Viewer database of the American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE). This database, which is discussed in Chapter 28 of the 2005 “ASHRAE Handbook—Fundamentals,” includes climatic design information for approximately 700 weather stations in the continental United States. The ASHRAE database includes statistics for each weather station, such as extreme wet-bulb, 0.4-percent annual exceedance wet-bulb, and 0.4-percent annual exceedance dry-bulb temperatures.

The ASHRAE extreme wet-bulb data represent hourly data (e.g., the highest of the values measured once each hour), whereas the AP1000 maximum safety coincident and noncoincident wet-bulb site parameter values of 30.1 °C (86.1 °F) exclude peaks of less than 2 hours. Consequently, the staff examined the ASHRAE database to identify those weather stations that had extreme wet-bulb data exceeding 30.6 °C (87.1 °F), assuming such occurrences would be equivalent to a 2-hour peak exceeding 30.1 °C (86.1 °F). The staff found that approximately 15 percent (97 out of 660) of the weather stations located throughout the continental United States had an extreme wet-bulb value exceeding 30.6 °C (87.1 °F). Because only a small number (i.e., 15 percent) of weather stations had an extreme wet-bulb value that exceeded 30.6 °C (87.1 °F), the staff concludes that the AP1000 maximum safety coincident and noncoincident wet-bulb air temperature site parameter values of 30.1 °C (86.1 °F) can be expected to bound a reasonable number of sites that have been or may be considered for a COL application.

The staff also examined the ASHRAE database to identify the number of weather stations that exceeded a 0.4-percent annual exceedance wet-bulb value of 26.7 °C (80.1 °F). The AP1000 maximum normal coincident and noncoincident wet-bulb site parameter values of 26.7 °C (80.1 °F) are 1-percent seasonal exceedance values, which are likely to be about the same as a 0.4-percent annual exceedance wet-bulb value of 26.7 °C (80.1 °F). The staff found that approximately 11 percent (75 out of 660) of the weather stations had a 0.4-percent wet-bulb value exceeding 26.7 °C (80.1 °F). Because only a small number (i.e., 11 percent) of weather stations had a 0.4-percent wet-bulb value that exceeded 26.7 °C (80.1 °F), the staff concludes that the AP1000 maximum normal coincident and noncoincident wet-bulb air temperature site parameter values of 26.7 °C (80.1 °F) can be expected to bound a reasonable number of sites that have been or may be considered for a COL application.

The staff also examined the ASHRAE database to identify the number of weather stations where the 0.4-percent annual exceedance dry-bulb value exceeded 38.3 °C (101 °F). The AP1000 maximum normal dry-bulb site parameter value of 38.3 °C (101 °F) is a 1-percent seasonal exceedance value that is likely to be about the same as a 0.4-percent annual exceedance dry-bulb value of 38.3 °C (101 °F). The staff found that approximately 5 percent (38 out of 700) of the weather stations had a 0.4-percent dry-bulb value exceeding 38.3 °C (101 °F). Because only a small number (i.e., 5 percent) of weather stations had a 0.4-percent dry-bulb value that exceeded 38.3 °C (101 °F), the staff concludes that the AP1000 maximum normal dry-bulb air temperature site parameter of 38.3 °C (101 °F) which is likely to bound a reasonable number of sites that have been or may be considered for a COL application.

2.3.1.1.5 Technical Conclusions

The applicant has selected a revised set of air temperature site parameters referenced above for plant design inputs, and the staff agrees that these revised site parameters can be expected to be representative of a reasonable number of sites that have been or may be considered for a COL application. This will ensure that GDC 2 is met, in that SSCs important to safety will be designed to withstand the effects of natural phenomena (e.g., extreme air temperatures) without losing the ability to perform their safety functions and will reduce the number of requests for exemptions or departures in future COL applications, which could occur if the FSAR cannot demonstrate that the design of the facility falls within the characteristics of the site.

AP1000 COL Information Item 2.3-1 states that COL applicants referencing the AP1000 design will address site-specific information related to regional climatology. The COL applicant will also need to demonstrate that the characteristics of the selected site fall within the site parameters specified in the design approval, pursuant to 10 CFR 52.79(c)(1). For a selected site with any of the air temperature site characteristics in excess of the corresponding AP1000 site parameters, the COL applicant will need to address how the SSCs important to safety will be able to withstand the effects of the natural phenomena without losing the ability to perform their safety functions in accordance with GDC 2.

In determining site characteristic values for comparison with the AP1000 maximum safety site parameter values, a COL applicant should select the higher of either: (1) the most severe value that has been historically reported for the site and surrounding area; or (2) the 100-year return period value. Regulations in 10 CFR 52.79(a)(1)(iii) state, in part, that the COL FSAR shall include the meteorological characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated. To comply with 10 CFR 52.79(a)(1)(iii), the maximum safety ambient temperature site-specific characteristic values identified by the COL applicant should be based on the higher of either: (1) the historic maximum values recorded in the site vicinity; or (2) the 100-year return period values. Temperatures based on a 100-year return period are considered to provide sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, as required by the regulation.

APP-GW-GLE-036 states that the revisions to the maximum safety coincident and noncoincident wet-bulb temperatures were implemented to encompass more sites in the eastern United States, such as Levy and Turkey Point. APP-GW-GLE-036 further states that Progress Energy chose the revised wet-bulb temperature values to support the COL application for the Levy site, to avoid any departures from the AP1000 design. The staff’s acceptance of the revised AP1000 maximum safety coincident and noncoincident wet-bulb temperature values as being expected to bound a reasonable number of sites does not imply that the staff finds that these revised values bound the corresponding site characteristic values for any given COL site, such as the Levy site. The staff will assess the maximum safety coincident and noncoincident wet-bulb temperature site characteristic values as part of its review of a COL application.

2.3.1.2 Conclusion

The staff has reviewed the information presented by the applicant and concludes that the changes in air temperature site parameters are acceptable, because they meet the requirements of GDC 2 in Appendix A to 10 CFR Part 50 and 10 CFR 52.63(a)(1), “Finality of standard design certifications,” as well as the associated acceptance criteria specified in NUREG-0800 Section 2.3.1.

2.3.4 Short-Term (Accident) Atmospheric Relative Concentration

Revision 17 to the AP1000 DCD made changes to some of the control room (CR) atmospheric dispersion factors (also known as atmospheric relative concentration or χ/Q values) presented in DCD Revision 15. The staff benchmarked the Revision 17 changes against Revision 15, which is the previously staff-approved version of the AP1000 DCD. The applicant made the following changes:

- The applicant revised the CR χ/Q values presented in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Tables 2-1 and 15A-6, for plant vent or passive containment cooling system (PCS) air diffuser and ground-level containment releases to the CR heating, ventilation, and air conditioning (HVAC) intake and annex building door. Table 2.3.4-1 of this SER lists these revisions.

- The applicant added CR χ/Q values for condenser air removal stack releases to the HVAC intake and annex building door to DCD Tier 1, Table 5.0-1, and DCD Tier 2, Tables 2-1 and 15A-6. SER Table 2.3.4-1 presents a list of these revisions.

- The applicant revised some of the CR source and receptor data provided in DCD Tier 2, Table 15A-7, for determining CR atmospheric dispersion factors. SER Table 2.3.4-2 lists these revisions.

The following served as source documents for these revisions:

- APP-GW-GLE-001 Revision 0, March 7, 2008, “Impact of Annex Building Expansion and Condenser Air Removal Stack Location on the Control Room Atmospheric Dispersion Factors”

- APP-GW-GLN-122 Revision 0, July 2007, “Offsite and Control Room Dose Changes”

2.3.4.1 Evaluation

The staff prepared SER Section 2.3.4 in accordance with the review procedures described in NUREG-0800 Section 2.3.4, using information presented in Revision 17 of the AP1000 DCD, APP-GW-GLE-001, APP-GW-GLN-122, and the applicant’s responses to RAIs on APP-GW-GLE-001 and APP-GW-GLN-122. Where appropriate, the applicant has incorporated its RAI responses in Revision 17 of the DCD. Since the staff has reviewed the DCD Revision 17 and DCD Revision 17 includes the incorporation of the RAI responses, the staff considers the RAIs related to the DCD to be closed.

2.3.4.1.1 General Description

Section 2.3.4 addresses, among other items, the χ/Q estimates at the CR for postulated design-basis accidental radioactive airborne releases. In lieu of site-specific meteorological data, the applicant provided a set of hypothetical, short-term CR χ/Q values to evaluate the AP1000 design. The set of AP1000 site parameters listed in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, includes these CR χ/Q values. DCD Tier 2, Section 2.3.4, states that the applicant derived the short-term χ/Q site parameters from a study performed to determine the short-term χ/Q values that would envelop most current plant sites. The CR radiological consequence analyses presented in DCD Tier 2, Sections 6.4 and 15.6.5, use the resulting CR short-term χ/Q values.

2.3.4.1.2 Description of Proposed Changes

(1) Changes in Plant Vent or PCS Air Diffuser and Ground-Level Containment Release χ/Q Values

SER Table 2.3.4-1 lists the applicant’s changes to the CR χ/Q values from DCD Revision 15 to DCD Revision 17 for plant vent or PCS air diffuser and ground-level containment releases to the HVAC intake and annex building door. SER Table 2.3.4-1 shows that all plant vent or PCS air diffuser and ground-level containment release CR χ/Q values increased in DCD Revision 17. The extent of this increase ranged from 36 percent to over 400 percent.

The CR habitability analyses used the HVAC intake χ/Q values for: (a) evaluating the time period preceding the isolation of the main CR and actuation of the emergency habitability system; (b) evaluating the time period after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main CR; and (c) determining CR doses when the nonsafety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated. The analyses used the annex building door χ/Q values when the emergency habitability system is in operation and the only pathway for contaminated air entering the CR is assumed to be the result of ingress or egress.

The applicant's source document for these revisions in atmospheric dispersion factors is APP-GW-GLN-122. Revision 0 to this document described three changes implemented in the AP1000 DCD, Revision 16 that reduced some of the calculated radiological doses off site and in the main CR for design-basis accidents. These three changes were: (a) directing the main CR emergency habitability system discharge airflow into the entry vestibule to provide a continuous vestibule purge; (b) increasing the decay time in Technical Specification 3.9.7, "Decay Time, Refueling Operations," from 24 hours to 48 hours to provide increased radioactive decay of short-lived fission products before irradiated fuel assemblies are handled; and (c) revising the calculation of radioactivity released for the postulated loss-of-coolant accident (LOCA) to take credit for aerosol impaction removal in the containment leakage pathway. The staff approved the first two changes but did not approve the last change; nonetheless, the first two changes allowed the CR atmospheric dispersion site parameter values shown in SER Table 2.3.4-1 to be increased to accommodate sites with higher χ/Q values than those originally specified in the AP1000 DCD, Revision 15. Larger χ/Q values are associated with less dilution capability, resulting in higher radiological doses. When comparing a site parameter χ/Q value and a site characteristic χ/Q value, the site is acceptable for the design if the site characteristic χ/Q value is smaller than the site parameter χ/Q value. Such a comparison shows that the site has better dispersion characteristics than those required by the reactor design.

(2) New Condenser Air Removal Stack Release χ/Q Values

SER Table 2.3.4-1 lists the new condenser air removal stack release χ/Q values presented in the AP1000 DCD, Revision 17. DCD Revision 15 did not present CR χ/Q values for this release pathway.

The applicant's source document for these new χ/Q values is APP-GW-GLE-001. This report addresses concerns associated with a correction made to the location of the condenser air removal stack, as shown in DCD Tier 2, Table 15A-7 and Figure 15A-1. The corrected location decreased the distance between the condenser air removal stack and the annex building access door. Footnote 5 in Revision 15 of DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, stated that the listed χ/Q values for the power-operated relief valve (PORV) and safety valve releases bound the dispersion factors for releases from the condenser air removal stack. With the revised location of the condenser air removal stack, the applicant was concerned that this statement may no longer be valid. Consequently, in APP-GW-GLE-001, the applicant: (a) modified Footnote 5 to eliminate the assertion that the listed χ/Q values for the PORV and safety valve releases bound the dispersion factors for releases from the condenser air removal stack; (b) added atmospheric dispersion factors specifically for the condenser air removal stack release point; and (c) added Footnote 7 to DCD Tier 1, Table 5.0-1, and DCD Tier 2, Tables 2-1 and 15A-6, which states that the condenser air removal stack release point was included for information only as a potential activity release point and none of the design-basis accident radiological consequence analyses model releases from this release point.

APP-GW-GLE-001 states that because the straight-line distances are similar, the applicant chose the same atmospheric dispersion factors for the condenser air removal stack releases to the HVAC intake as those currently defined values used for the release-receptor pair of the fuel-handling area to the HVAC intake. Similarly, APP-GW-GLE-001 states that, because the straight-line distances are similar, the applicant chose the same atmospheric dispersion factors for the condenser air removal stack releases to the annex building entrance as those currently defined values used for the release-receptor pair of PORV and safety values to the HVAC intake.

(3) Revised Control Room Source and Receptor Data

SER Table 2.3.4-2 lists the changes in CR source and receptor data between the AP1000 DCD, Revision 15 and the DCD, Revision 17. SER Table 2.3.4-2 shows that the horizontal straight-line distances from all release points (except for the condenser air removal stack) to the HVAC intake and annex building access receptors increased.

The applicant used APP-GW-GLE-001 as the source document for these source and receptor changes. This report addresses the impact of a relocation of the annex building entrance and HVAC intake on the CR source and receptor data to be used in determining site-specific CR χ/Q values. With an exception for the condenser air removal stack, the relocation of these two CR receptor locations increased the distances between the previously identified release points and these receptors. A correction made to the location of the condenser air removal stack, as discussed above, decreased the distances between the condenser air removal stack release pathway and the HVAC intake and annex building access receptors.

2.3.4.1.3 Applicable Regulations and Associated Acceptance Criteria

Acceptance criteria regarding the CR χ/Q site parameter values are based on meeting the relevant requirements of GDC 19, “Control Room,” in Appendix A of 10 CFR Part 50, which states, in part, that a CR shall be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including a LOCA. Atmospheric dispersion factors are an important component of the CR radiological habitability analyses used to demonstrate that the CR operator dose criterion in GDC 19 is met.

NUREG-0800 Section 2.3.4 states that the DC application should include CR atmospheric dispersion factors for the appropriate time periods in the list of site parameters. The DC application should also include figures and tables showing the design features that the COL applicant will use to generate CR χ/Q values (e.g., intake heights, release heights, building cross-sectional areas, and distance to receptors). NUREG-0800 Section 2.3.4 also states that the postulated site parameters should be representative of a reasonable number of sites that may be considered within a COL application and a basis should be provided for each of the site parameters. RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” presents criteria for characterizing atmospheric dispersion conditions for evaluating the consequences of radiological releases to the CR. RG 1.194 states that the ARCON96 atmospheric dispersion model (Revision 1 to NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes”) is an acceptable methodology for assessing CR χ/Q values for use in CR design-basis accident radiological analyses, subject to the provisions in RG 1.194.

2.3.4.1.4 Evaluation

This SER section is limited to reviewing the appropriateness of the values chosen as atmospheric dispersion site parameters; other SER sections (e.g., Sections 6.4 and 15.3) review the effects of the implemented χ/Q revisions on the design-basis dose calculations.

To confirm that the revised set of plant vent or PCS air diffuser and ground-level containment release CR χ/Q site parameters and the new set of condenser air removal stack release CR χ/Q site parameters presented in Revision 17 to the DCD are representative of a reasonable number of sites that have been or may be considered for a COL application, the staff generated site-specific χ/Q values for the four docketed early site permit (ESP) applications (North Anna, Clinton, Grand Gulf, and Vogtle) using the ARCON96 computer code with: (1) the revised source and receptor information presented in DCD Tier 2, Table 15A-7 (assuming the AP1000 plant north was aligned to true north at each site), and (2) the site-specific hourly meteorology data sets provided in support of each ESP application. The staff found that the AP1000 CR χ/Q site parameter values were bounding in all cases. Consequently, the staff finds that the applicant has provided CR atmospheric dispersion site parameter values that bound several sites that may be considered within a COL application and are, therefore, acceptable. The CR atmospheric dispersion site parameters will help to ensure that the CR operator dose criterion in GDC 19 is met. APP-GW-GLE-001 revised the CR χ/Q source and receptor data presented in DCD Tier 2, Table 15A-7, based on a correction made to the location of the condenser air removal stack and relocation of the annex building entrance and CR air inlet. In all cases (except for the condenser air removal stack), the distances between the sources and receptors increased. Since χ/Q values generally decrease as downwind travel distances increase, APP-GW-GLE-001 was conservative in that it did not change the CR atmospheric dispersion factors presented in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Tables 2-1 and 15A-6, to reflect the increases in downwind distances. The applicant based the revisions in χ/Q values presented in SER Table 2.3.4-1 on the changes implemented in response to the findings of APP-GW-GLN-122 as discussed previously. Based on the information above the staff finds this acceptable.

Table 2.3.4-1. Revisions to CR Atmospheric Dispersion Factor (χ/Q) Site Parameter Values (s/m^3)

SITE PARAMETER	DCD REVISION 15	DCD REVISION 17	% INCREASE
Plant Vent or PCS Air Diffuser Release to the HVAC Intake			
0–2 hours	2.2×10^{-3}	3.0×10^{-3}	136%
2–8 hours	1.4×10^{-3}	2.5×10^{-3}	179%
8–24 hours	6.0×10^{-4}	1.0×10^{-3}	167%
1–4 days	4.5×10^{-4}	8.0×10^{-4}	178%
4–30 days	3.6×10^{-4}	6.0×10^{-4}	167%
Plant Vent or PCS Air Diffuser Release to the Annex Building Door			
0–2 hours	6.6×10^{-4}	1.0×10^{-3}	152%
2–8 hours	4.8×10^{-4}	7.5×10^{-4}	156%
8–24 hours	2.1×10^{-4}	3.5×10^{-4}	167%
1–4 days	1.5×10^{-4}	2.8×10^{-4}	187%
4–30 days	1.3×10^{-4}	2.5×10^{-4}	192%
Ground-Level Containment Release to the HVAC Intake			
0–2 hours	2.2×10^{-3}	6.0×10^{-3}	273%
2–8 hours	1.4×10^{-3}	3.6×10^{-3}	257%
8–24 hours	6.0×10^{-4}	1.4×10^{-3}	233%
1–4 days	4.5×10^{-4}	1.8×10^{-3}	400%
4–30 days	3.6×10^{-4}	1.5×10^{-3}	417%
Ground-Level Containment Release to the Annex Building Door			
0–2 hours	6.6×10^{-4}	1.0×10^{-3}	152%
2–8 hours	4.8×10^{-4}	7.5×10^{-4}	156%
8–24 hours	2.1×10^{-4}	3.5×10^{-4}	167%
1–4 days	1.5×10^{-4}	2.8×10^{-4}	187%
4–30 days	1.3×10^{-4}	2.5×10^{-4}	192%

Table 2.3.4-1. Revisions to CR Atmospheric Dispersion Factor (χ/Q) Site Parameter Values (s/m^3)

SITE PARAMETER	DCD REVISION 15	DCD REVISION 17	% INCREASE
Condenser Air Removal Stack Release to the HVAC Intake 0–2 hours 2–8 hours 8–24 hours 1–4 days 4–30 days	None Provided	 6.0×10^{-3} 4.0×10^{-3} 2.0×10^{-3} 1.5×10^{-3} 1.0×10^{-3}	--
Condenser Air Removal Stack Release to the Annex Building Door 0–2 hours 2–8 hours 8–24 hours 1–4 days 4–30 days	None Provided	 2.0×10^{-2} 1.8×10^{-2} 7.0×10^{-3} 5.0×10^{-3} 4.5×10^{-3}	--

Table 2.3.4-2. Revisions to CR Atmospheric Dispersion Factor (χ/Q) Site Parameter Values (s/m^3)

RELEASE POINT	RELEASE ELEVATION		HORIZONTAL STRAIGHT-LINE DISTANCE TO RECEPTOR			
			HVAC INTAKE (ELEVATION 19.9 METERS (m))		ANNEX BUILDING ACCESS (ELEVATION 1.5 METERS (m))	
	REVISION 15	REVISION 17	REVISION 15	REVISION 17	REVISION 15	REVISION 17
Plant Vent	55.7 m	No Change	39.6 m	44.9 m	76.8 m	115.6 m
PCS Air Diffuser	71.3 m	69.8 m	32.3 m	36.0 m	68.9 m	104.6 m
Fuel Building Blowout Panel	17.4 m	No Change	50.0 m	61.9 m	89.7 m	130.3 m
Fuel Building Rail Bay Door	1.5 m	No Change	52.4 m	66.6 m	92.1 m	132.1 m
Steam Vent	17.1 m	No Change	18.3 m	18.8 m	48.8 m	79.7 m
PORV/Safety Valves	19.2 m	No Change	19.8 m	20.4 m	44.1 m	77.8 m
Condenser Air Removal Stack	7.6 m	38.4 m	63.0 m	60.4 m	59.9 m	17.8 m
Containment Shell	Same as receptor elevation (19.9 m or 1.5 m)	No Change	11.0 m	12.8 m	47.2 m	83.0 m

2.3.4.1.5 Technical Conclusions

The applicant has selected a revised set of short-term (accident) CR atmospheric dispersion site parameters referenced above for plant design inputs. The staff agrees that these revised CR χ/Q values can be expected to be representative of a reasonable number of sites that have been or may be considered for a COL application. AP1000 COL Information Item 2.3-4 states, in part, that a COL applicant referencing the AP1000 design will address the site-specific CR χ/Q values. For a site selected that exceeds the bounding CR χ/Q values, COL Information Item 2.3-4 further states that the COL applicant will address how the radiological consequences associated with the controlling design-basis accident continue to meet the CR operator dose limits given in GDC 19 using site-specific χ/Q values. The staff concludes that successful completion of COL Information Item 2.3-4 will demonstrate that the short-term (accident) atmospheric dispersion factors for the CR will be acceptable.

2.3.4.2 Conclusion

The staff has reviewed the information presented by the applicant and concludes that the changes in short-term (accident) CR site parameters are acceptable because they meet the requirements of GDC 19 and 10 CFR 52.63(a)(1) and the associated acceptance criteria specified in NUREG-0800 Section 2.3.4.

2.4 Hydrologic Engineering

2.4.1 Hydrological Description

The AP1000 is a standard design with a plant configuration that assumes a normal water level at 0.6 meters (m) (2 feet (ft)) below the grade, and a flood level at the design plant grade of 30.5 m (100 ft). The actual grade level will be a few inches lower to prevent surface water ingress through the doorways. This provision recognizes that the Utility Requirements Document (URD) states that the maximum flood (or tsunami) level site envelope parameter is 0.3 m (1 ft) below grade. Although the AP1000 design flood level of 30.5 m (100 ft) does not meet the URD flood level criterion explicitly, this deviation is considered inconsequential to safety.

The maximum flood level mentioned above is based on a site parameter referred to as the probable maximum flood (PMF). The PMF is the flood that may be expected from the most severe combination of critical meteorological and hydrologic conditions that are reasonably possible in a particular drainage area and is generated by a separate parameter called the probable maximum precipitation (PMP). The PMP is the greatest depth (amount) of precipitation, for a given storm duration, that is theoretically possible for a particular area and geographic location. PMP values are typically found in the National Weather Service hydro-meteorological reports (HMRs).

The applicant proposed a change to the PMP parameter value from 0.0137 centimeters/second (cm/s) (19.4 inches per hour (in/h)) to 0.0146 cm/s (20.7 in/h) in the AP1000 DCD, Revision 17.

2.4.2 Regulatory Basis

The staff considered the following regulatory requirements in reviewing the applicant's submittal:

- 10 CFR 100.20(c)(3), as it relates to the PMF

- 10 CFR 52.47(a)(1), as it relates to the site parameters postulated for the design

- 10 CFR 52.79(a)(1)(iii), as it relates to the hydrologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated

- GDC 2, which states in part that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of capability to perform their safety functions

2.4.3 Summary of Technical Information

In Revision 0 of APP-GW-GLE-012, "Probable Maximum Precipitation Value Increase," the applicant proposed to change the PMP value from 0.0137 cm/s (19.4 in/h) to 0.0146 cm/s (20.7 in/h). This value is found in Tier 1, Table 5.0-1, "Site Parameters," on page 5.0-2, and in Tier 2, Table 2-1 (Sheet 3 of 4), "Site Parameters," on page 2-21 of the AP1000 DCD, Revision 17.

2.4.4 Evaluation

The applicant has determined a new PMP value of 0.0146 cm/s (20.7 in/h) based on an interpretation of Figure 24 in HMR-52 from the National Weather Service. The staff, while not agreeing with this interpretation of Figure 24 found in HMR-52, does agree with the applicant's statements made in the associated AP1000 DCD impact document and has no objection to this change in the PMP value for the AP1000 DCD. The staff held a phone conference call with the applicant on August 21, 2008, to discuss technical issues related to the change. As a follow-up to that phone call, the staff issued RAI-SRP2.4-RHEB-01. The RAI included three surface water and three ground water questions. The first surface water question was associated with Table 3.3-5, Tier 1, inspections, tests, analyses, and acceptance criteria (ITAAC) Design Commitment 2.b related to the tolerance value of ±1.07 m (±3.5 ft) between the design plant grade and the site grade. In a letter dated September 15, 2008, the applicant responded to RAI-SRP2.4-RHEB-01. Specifically, the applicant, in its response to this question, stated that the tolerance of 1.07 m (3.5 ft) between design plant grade and site grade in DCD Tier 1, Table 3.3-5, is based on seismic and soil-structure interaction (SSI) considerations for the auxiliary, shield, and containment buildings. Furthermore, this tolerance is not related to hydrology or surface water considerations. The applicant further stated that it is not appropriate to use this tolerance to establish the relationship between the design plant grade and the PMF. Based on this clarification, the staff finds the response acceptable and considers this question resolved.

The second surface water question asked the applicant to specify where on the site the ITAAC Design Commitment 2.b should be met and to which buildings the commitment should be applied. In the September 15, 2008, letter, the applicant stated that the zone of influence of soil characteristics on the structural response of an embedded structure is generally considered to extend horizontally away from the structure the same distance as the depth of the embedment.

For the AP1000, this distance is approximately 12.2 m (40 ft) from the auxiliary and shield buildings. Additionally, the applicant stated that other evaluations and analyses address the effects of buildings founded at grade adjacent to the nuclear island on the seismic interaction. The applicant also stated that ITAAC Commitment 2.b in DCD Tier 1, Section 3.3, does not apply to site surface water flooding. Based on this information, the staff considers the applicant's response to be acceptable, and the issue is resolved.

The third surface water question asked the applicant to describe the expected vertical distance and tolerance between: (1) the design plant grade; (2) the to-be-built site grade; and (3) the maximum surface water elevation associated with a flood (see Table 5.0-1, DCD Tier 1) and to identify to which building these distances and tolerances apply. In the September 15, 2008, letter, the applicant stated that Table 5.0-1 includes the COL information specifying the compliance of the site PMF level with the plant site design parameters. This table defines the distance between the design plant grade of elevation 30.5 m (100 ft) and the maximum surface water elevation. The applicant also stated that ITAAC Commitment 2.b in DCD Tier 1, Section 3.3, does not define the distance between the design plant grade of elevation 30.5 m (100 ft) and the maximum surface water elevation. The staff finds this response acceptable and considers this issue resolved.

The first ground water question in RAI-SRP2.4RHEB-01 asked the applicant to clarify its definition of normal ground water elevation in Tier 2 of the DCD. In the September 15, 2008, letter, the applicant stated that Table 5.0-1 of DCD Tier 1 defines the maximum ground level as plant elevation 98 ft and the maximum flood level as plant elevation 30.5 m (100 ft.) The applicant also stated that the reference to normal ground water is applicable at all times except when there is surface water flooding. The staff found this response to be unacceptable because the applicant did not specify the maximum ground water level, but instead allowed an exception to the ground water level under certain conditions. This issue was Open Item OI-SRP2.4RHEB-01-01. In its response to RAI-SRP2.4RHEB-01, the applicant retracted the statement referencing normal ground water levels except under conditions of surfacing water flooding and made clear there are no exceptions to the normal ground water elevation. With this exception removed, this response is acceptable to the staff, and Open Item OI-SRP2.4RHEB-01-01 is resolved.

The second ground water question in RAI-SRP2.4RHEB-01 asked the applicant to specify to which buildings in Table 5.0-1, DCD Tier 1 the maximum ground water level elevations should be applied. The applicant replied that the DCD Tier 1, Table 5.0-1, specification of maximum flood level at plant elevation 30.5 m (100 ft) (design-grade elevation) is specifically applicable to the safety-related nuclear island. Furthermore, the buildings adjacent to the nuclear island are founded at grade and use the same reference elevation designation as the auxiliary building and the containment building. The applicant also stated that differences in actual elevation between the nuclear island and the adjacent buildings conform to standard construction tolerances and are independent of site grade variation.

The applicant further stated that the site grading, including local slope to encourage run off away from the doorways of the buildings included in the certified design, is site-specific. Based on the information, the staff finds this response acceptable, and the issue is resolved.

The third ground water question in RAI-SRP2.4RHEB-01 asked the applicant to specify the maximum allowed water table elevation and the maximum time this elevation can be sustained without an increase in safety risk. The applicant responded stating that the normal water table elevation is expected to be exceeded only during surface water flooding events. In addition, while surface water flooding may impede access to the AP1000, the AP1000 is designed to cope with impeded access for a period of 7 days. The staff found this response unacceptable because the applicant failed to specify the maximum allowed water table and the time this elevation can be sustained without an increase in safety risk. This issue was Open Item OI-SRP2.4RHEB-01-02. In response to RAI-SRP2.4RHEB-01, the applicant retracted the statement referencing normal ground water levels except under conditions of surface water flooding and made clear there are no exceptions to the normal ground water elevation. With the removal of this exception, this response is acceptable to the staff and Open Item OI-SRP2.4RHEB-01-02 is resolved.

2.4.5 Conclusion

The applicant has presented information relative to the PMP value found in AP1000 DCD Tier 1, Table 5.0-1, and in DCD Tier 2, Table 2-1 (Sheet 3 of 4). The staff reviewed the information provided and considers all RAIs and open items to be resolved. Additionally, the staff concludes that this portion of the application meets the requirements of GDC 2, 10 CFR Part 52, “Licenses, certifications, and approvals for nuclear power plants,” and 10 CFR Part 100, relating to hydrologic characteristics.

2.5 Geological, Seismological, and Geotechnical Engineering

In Section 2.5, “Geology, Seismology, and Geotechnical Engineering,” of Revision 17 of the AP1000 DCD, Tier 2, the applicant described geologic, seismic, and geotechnical engineering properties required for a COL applicant referencing this standard design. DCD Section 2.5.1, “Basic Geologic and Seismic Information,” presents geologic and seismic characteristics of the site and region that COL applicants referencing the AP1000 DCD need to address. DCD Section 2.5.2, “Vibratory Ground Motion,” identifies the vibratory ground motion assessment, including the safe-shutdown earthquake (SSE) and design response for the COL applicant to follow. DCD Section 2.5.3, “Surface Faulting Combined License Information,” describes the requirements for the COL applicant to address regarding the potential for surface tectonic and nontectonic deformation. DCD Sections 2.5.4, “Stability and Uniformity of Subsurface Materials and Foundations,” and 2.5.5, “Combined License Information for Stability and Uniformity of Slopes,” describe the foundation and subsurface material stability criteria to be met by COL applicants. DCD Section 2.5.6, “Combined License Information for Embankments and Dams,” discusses requirements for stability of embankments and dams near the COL site.

The six main sections of this part of the SER (i.e., Section 2.5) parallel the six main sections included in the applicant’s DCD. Except for the sections where the applicant made no changes from Revision 15 of the AP1000 DCD, the SER sections are divided into six sections: (1) the “Introduction” section, which briefly describes the contents of each main DCD section; (2) the “Technical Information in the Application” section, which describes the technical content of the DCD; (3) the “Regulatory Basis” section, which summarizes the regulations and NRC regulatory guides used by the staff to review the DCD; (4) the “Evaluation” section, which describes the staff’s evaluation of what the applicant did, including requests for RAIs and open items, and confirmatory analyses performed by the staff, if applicable; (5) the “Post Combined License Activities” section, which identifies related post-COL activities; and (6) the “Conclusions” section, which provides the staff’s conclusions and documents whether the applicant provided sufficient and adequate information to meet all relevant regulatory requirements.

The staff also reviewed the AP1000 DCD Tier 1 information that is related to DCD Tier 2, Section 2.5, and incorporated the Tier 1 information review into the appropriate sections of the Tier 2 DCD review discussed in this SER section. The SER focuses on the changes the applicant made in Revision 17 of the AP1000 DCD as compared to the previously certified revision of the DCD.

2.5.1 Basic Geologic and Seismic Information

The applicant made no changes or additions to DCD Section 2.5.1 from Revision 15 of the AP1000 DCD. Therefore, the staff did not reevaluate any of the previously certified information included in this section.

2.5.2 Vibratory Ground Motion

2.5.2.1 Introduction

DCD Section 2.5.2 states that the AP1000 certified seismic design response spectra (CSDRS) were developed using the response spectra of RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” as the base. The applicant then modified the base spectra to include additional high-frequency amplification at a control point at 25 Hertz (Hz) with equal peak ground acceleration (PGA) in the horizontal and the vertical directions, as presented in Figures 3.7.1-1 and 3.7.1-2 in the DCD. The applicant also stated that for a site, at which the nuclear island is founded on hard rock, the design response spectra specified in Appendix 3I to the DCD and Figures 3I.1-1 and 3I.1-2 can be used in place of the CSDRS.

2.5.2.2 Technical Information in the Application

2.5.2.2.1 Combined License Seismic and Tectonic Characteristics Information

AP1000 DCD, Section 2.5.2.1, “Combined License Seismic and Tectonic Characteristics Information,” states that the site-specific ground motion response spectra (GMRS) would be defined at the ground surface in the free-field and compared to the CSDRS. For sites with soil layers that will be completely excavated to expose competent material (in situ material with a shear wave velocity of 305 m/s (1000 feet per second (fps)) or higher), the applicant stated that the GMRS will be specified on an outcrop or a hypothetical outcrop that would exist after excavation. The applicant further clarified that the motions at the hypothetical outcrop are developed as a free-surface motion, not as an in-column motion with no soil above the outcrop.

In addition, the applicant described seven requirements in AP1000 DCD Section 2.5.2.1 for the COL applicant to address in order to demonstrate that a selected site was suitable for the AP1000 standard design. The applicant updated the following five requirements in Revision 17 of the DCD:

- For a site at which the nuclear island is founded on hard rock with a shear wave velocity greater than 2,438 m/s (8,000 fps), the site-specific GMRS can be defined at the foundation level and may be shown to be less than or equal to the CSDRS.

- For a site at which the nuclear island is directly founded on hard rock, the site-specific PGA and spectra should be developed for the top of competent rock and shown to be less than or equal to those values given in DCD Figures 3I.1-1 and 3I.1-2 at the foundation level and over the entire frequency range.

- Layers of the soil beneath the foundation are approximately horizontal, sloping less than 20 degrees, and the minimum estimate of the low-strain shear wave velocity of the soil underneath the nuclear island foundation is greater than or equal to 305 m/s (1,000 fps).

- For sites at which the nuclear island is founded on soil, the median estimate of the strain-compatible soil shear modulus and hysteretic damping is compared to the values used in the AP1000 generic analyses shown in DCD Table 3.7.1-4 and Figure 3.7.1-17. Properties of soil layers within a depth of 36.6 m (120 ft) below finished grade are compared to those in the generic soil site analyses (soft soil (SS), soft-to-medium (SM) soil, and upper bound soft-to-medium (UBSM) soil). The shear wave velocity should also increase with depth, and the average low-strain shear wave velocity should not be less than 80 percent of the average shear wave velocity at a higher elevation.

- A site-specific evaluation, as described in DCD Section 2.5.2.3, may be performed in lieu of the other requirements.

DCD Tier 1, Table 5.0-1, specifies the site parameter for the SSE as follows:

> SSE free-field peak ground acceleration of 0.30 g with modified regulatory guide 1.60 response spectra (See Figures 5.0-1 and 5.0-2). Seismic input is defined at finished grade except for sites where the nuclear island is founded on hard rock. If the site-specific spectra exceed the response spectra in Figures 5.0-1 and 5.0-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed. This evaluation will consist of a site-specific dynamic analysis and generation of in-structure response spectra at key locations to be compared with the floor response spectra of the certified design at 5-percent damping. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations or the exceedances are justified.
>
> The hard rock high frequency (HRHF) ground motion response spectra (GMRS) are shown in Figure 5.0-3 and Figure 5.0-4 defined at the foundation level for 5 percent damping. The HRHF GMRS provides an alternative set of spectra for evaluation of the site-specific GMRS. A site is acceptable if its site-specific GMRS falls within the AP1000 HRHF GMRS.

Revision 17 of the DCD added Figures 5.0-1 and 5.0-2 in Tier 1, Section 5.0, accordingly.

DCD Tier 1, Table 5.0-1, also states that there should be no potential for fault motion in the site area.

2.5.2.2.2 Site-Specific Seismic Evaluation

In DCD Tier 2, Section 2.5.2.3, “Site-Specific Seismic Evaluation,” the applicant revised the requirements to clarify that, if the site-specific spectra at foundation level exceed the response spectra in Figures 3.7.1-1 and 3.7.1-2 at any frequency, or if soil conditions were outside the range evaluated for AP1000 DC, a site-specific evaluation can be performed. For sites at which the response spectra exceed the CSDRS, or at which the soil parameters are outside those specified in the DCD, the applicant concluded that either a two-dimensional (2-D) or three-dimensional (3-D) site-specific analysis can be used to demonstrate site suitability.

Two-Dimensional Analyses

The applicant stated that for those features that were not within the site parameters, a site-specific SSI analysis may be performed following the guidance in Appendix 3G to the AP1000 DCD. The applicant stated that the results of such an analysis would need to be compared with the results of the 2-D SASSI analyses described in Appendix 3G and should demonstrate that local features are within the bounds established in the DCD. If the 2-D results are not clearly enveloped at significant frequencies of response, the applicant concluded that a 3-D analysis might be required.

Three-Dimensional Analyses

The applicant described the 3-D analyses that may be required if the 2-D results are inconclusive. The 3-D analyses would consist of a site-specific dynamic analysis and generation of in-structure response spectra at six key locations. Upon completion of the analysis, the COL applicant will need to compare the results with the floor response spectra of the certified design at 5-percent damping. The applicant specified that the CSDRS should be used to develop the floor response spectra, and they should be applied at the foundation level for the hard rock site and at finished grade for a soil site. The applicant concluded that the site would be acceptable if the floor response spectra from the site-specific evaluation did not exceed the AP1000 spectra for each of the following locations: containment internal structures at elevation of reactor vessel support, containment operating floor, auxiliary building at northeast corner elevation of 35.5 m (116.5 ft), shield building at fuel building roof, shield building roof, and the steel containment vessel at polar crane support.

2.5.2.3 Regulatory Basis

The staff relied on the following applicable regulatory requirements and guidance in reviewing the applicant’s discussion of vibratory ground motion:

10 CFR 52.47, with respect to requiring COL applicant to provide site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters

- 10 CFR Part 50, Appendix A, GDC 2, as it relates to consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity and period of time in which the historical data have been accumulated

- 10 CFR 100.23, ”Geologic and seismic siting criteria,” with respect to obtaining geologic and seismic information necessary to determine site suitability and ascertain that any new information derived from site-specific investigations would not impact the GMRS derived by a probabilistic seismic hazard analysis

- RG 1.132, “Site Investigations for Foundations of Nuclear Power Plants”

- RG 1.206, “Combined License Applications for Nuclear Power Plants”

- RG 1.208, “A Performance-Based Approach to Define Site-Specific Earthquake Ground Motion”

2.5.2.4 Evaluation

The applicant stated in Section 2.5.2 that "the AP1000 is also evaluated for a safe shutdown earthquake defined by a peak ground acceleration of 0.30 g and the design response spectra specified in Appendix 3I and Figures 3I.1-1 and 3I.1-2. These design response spectra are applicable to certain east coast rock sites." After examining DCD Figures 3I.1-1 and 3I.1-2, the staff asked the applicant, in RAI-SRP2.5-RGS1-01, to clarify what kind of response spectra the figures presented: GMRS or CSDRS, and to explain why the figures showed a PGA of 0.25 g.

In response to the RAI, the applicant revised the DCD text to clarify that Figures 3I.1-1 and 3I.1-2 showed HRHF response spectra resulting from the applicant's evaluations of hard rock sites, as described in Appendix 3I to the DCD. The applicant clarified that HRHF is not the design spectra, but it is the response spectra that can be used to evaluate the hard rock sites when the site-specific GMRS exceed the CSDRS shown in DCD Figures 3.7.1-1 and 3.7.1-2. The applicant stated that if the site-specific spectra are enveloped by the HRHF, it is non-damaging, and that AP1000 CSDRS control the AP1000 design. The details of the staff's evaluation of the applicant's process to determine the HRHF spectra are described in SER Section 3.7.1.

The applicant also revised Section 2.5.2 of the AP1000 DCD in response to this RAI to state that the AP1000 was designed for an earthquake with a PGA of 0.30 g, referring to the AP1000 CSDRS. In its response, the applicant explained that the PGA of 0.25 g addressed in RAI-SRP2.5-RGS1-01 is not that of the CSDRS, but it is the PGA of the HRHF spectra described above. The staff considers this response adequate as it clarifies the differences between the AP1000 CSDRS and the HRHF shown in DCD Figures 3I.1-1 and 3I.1-2 and the differences between the PGA values of the CSRDS and the HRHF spectra. Since the applicant revised the DCD to clarify the roles of CSDRS and HFRS, the staff considers RAI-SRP2.5-RGS1-01 resolved.

2.5.2.4.1 Combined License Seismic and Tectonics Characteristics Information

The staff considered the guidance in NUREG-0800 while reviewing the use of backfill soil to support the seismic Category I structures. In RAI-SRP2.5-RGS1-02, the staff asked the applicant to clarify how the GMRS would be calculated when backfill soil was involved. In response to this RAI, the applicant revised the DCD to clarify that no soil or backfill layers may exist above the outcrop when determining a site-specific GMRS. The staff reviewed this update in Revision 17 of the AP1000 DCD and noted that the revised DCD clearly describes how the site-specific GRMS should be determined. Since the revised DCD text clearly states that GMRS calculations will not include an overlying soil column, the staff concludes that the applicant's revised DCD satisfies the regulatory requirements; thereby, the staff considers RAI-SRP2.5-RGS1-02 resolved.

The staff found that, in general, requiring the COL applicant to demonstrate that the proposed site satisfies the seven requirements as described in the DCD meets NUREG-0800 guidelines; however, some issues needed to be clarified. In RAI-SRP2.5-RGS1-03, the staff asked the applicant to address the following issues of concern:

- Define "thin soil layer" and "soft soil layer" referred to in Requirement 4.

- Replace the phrase "median estimate" with the phrase "minimum estimate" in Requirement 5.

- Provide acceptance criteria and a basis to show the comparison to be acceptable in Requirement 6.

In response to this RAI, the applicant revised DCD Section 2.5.2.1 by eliminating the sentence containing "thin soil layer" and "soft soil layer" and replacing "median estimate" with "minimum estimate." The applicant also referred to detailed information regarding acceptance criteria for foundation soil in Section 3.7.1.4 of the DCD. After review of these revisions to the DCD, as well as the acceptance criteria for foundation soils found in Section 3.7.1.4 of the DCD, the staff concludes that this information is insufficient to resolve the issues identified in RAI-SRP2.5-RGS1-03 because the information does not satisfy the sixth screening requirement. The staff tracked this as Open Item OI-SRP2.5-RGS1-03.

To resolve the issues identified in Open Item OI-SRP2.5-RGS1-03, the applicant submitted a revised response dated November 9, 2009. In its response, the applicant proposed a revision to the DCD that would make the site acceptance criteria and the six screening criteria described in AP1000 DCD Tier 1 Section 2.5 consistent with those used in site response analyses, seismic system analyses, and SSI analyses. The most important site parameter is the shear wave velocity of the generic site soil profiles. The proposed DCD revision requires the shear wave velocities of the three generic soil profiles (SS, SM soil, and UBSM soil) to be within the lower and upper bounds of the shear wave velocities of the individual layers constituting the site-specific soil profiles. The lower bound and upper bound shear wave velocities correspond to Gmax/1.5 and 1.5*Gmax, respectively, where Gmax is the low-strain maximum shear modulus. The minimum shear wave velocity; however, will still be greater than or equal to 305 m/s (1000 fps). Since the applicant adequately addressed the concerns of the staff by making the site acceptance criteria consistent with the rest of the DCD, and committed to revise the DCD. In a subsequent revision to the AP1000 DCD, the applicant included these changes in the DCD text.

In Section 2.5.2.1 of the DCD, the applicant stated that, when site-specific parameters were not enveloped by the AP1000 standard design, a COL applicant might perform site-specific SSI analyses based on 2-D SASSI models and compare the results with those documented in Appendix 3G to DCD Chapter 3 to determine the adequacy of the standard design for the site. However, in Section 2.5.2.3 of DCD Revision 15, the applicant stated that site-specific SSI analyses should be performed using the 3-D SASSI models described in Appendix 3G. The staff asked the applicant, in RAI-SRP2.5-RGS1-04, to clarify the inconsistency and explain why the AP1000 DCD does not require the COL applicant to perform 3-D SSI analysis for a site at which 3-D effects cannot be ignored (such as a site with sloping excavation). In response to this RAI, the applicant moved the entire paragraph relating to the COL applicant's performance of site-specific SSI analysis from this section to DCD Section 2.5.2.3 and changed the section title from "Sites with Geoscience Parameters outside the Certified Design" to "Site Specific Evaluation." The applicant also explained that a COL applicant would perform a site-specific SSI analysis based on actual site conditions, and if a 2-D analysis was adequate the 3-D analysis would be unnecessary, as discussed in response to RAI-TR85-SEB1-07 and RAI-TR03-015. Furthermore, the applicant added Sections 2.5.2.3.1, "2-D Analyses," and 2.5.2.3.2, "3-D Analyses," to Revision 17 of the DCD. The staff considered these revisions of the AP1000 DCD and finds that, although the revised DCD added two separate sections to define when a 2-D or 3-D analysis would be required, it did not fully address the concerns of the staff described in RAI-SRP2.5-RGS1-04, RAI-TR85-SEB1-07 and in RAI-TR03-015, about the adequacy of a 2-D SSI analysis for an AP1000 structure where loads are not evenly applied on its foundation. The staff was concerned that the site-specific analysis should consider a 3-D effect for site conditions outside the certified design. This issue was tracked as Open Item OI-SRP2.5-RGS1-04.

In a letter dated December 9, 2009, the applicant addressed the staff's concerns described in Open Item OI-SRP2.5-RGS1-04. In its response, the applicant agreed to modify the DCD by adding a requirement that site-specific analysis should consider 3-D effects for cases where site parameters fall outside the certified design and loads are not evenly applied throughout the AP1000 foundation. The staff reviewed the response and concluded that the proposed revision of the AP1000 DCD provides adequate criteria for a site where the site parameters do not meet the certified design. Performing site-specific analyses with consideration of 3-D effects will ensure the stability of structures and foundations. In a subsequent revision to the AP1000 DCD, the applicant included these changes in AP1000 DCD, Tier 2 Section 2.5.2.3, and the issue is closed.

The staff reviewed APP-GW-GLE-004, Revision 0, "Soil and Seismic Parameter Change," with respect to shear wave velocity conditions and the statement made regarding minimum shear wave velocity. In RAI-SRP2.5-RGS1-15, Question 3, Issue 4, the staff asked the applicant to provide the criterion for the case of a soil layer with low-strain shear wave velocities of less than 762 m/s (2,500 fps). In Issue 5 of Question 3 of the same RAI, the staff also asked the applicant to revise the statement made regarding minimum shear wave velocity from "greater than or equal to 1000 fps based on low-strain, best estimate soil properties over the footprint of the nuclear island at its excavation depth" to "greater than or equal to 305 m/s (1000 fps) based on low-strain, minimum soil properties at its excavation depth."

In its response to RAI-SRP2.5-RGS1-15, the applicant first explained that Revision 15 of the AP1000 DCD originally included the criterion for the low-strain shear wave velocity of less than 762 m/s (2,500 fps), but the criterion was removed as indicated in APP-GW-GLE-004. The applicant explained that the tight limits of ±10 percent stated in the previous revision of the DCD were found to be unrealistic based on shear wave velocity variability. The applicant concluded that soil sites would require site-specific evaluation rather than following some special case. With respect to Issue 5, the applicant responded by stating that it would revise DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, to reflect the criterion for the minimum shear wave velocity.

The staff reviewed the applicant's responses, and determined that elimination of the criterion for soil layers with seismic velocities less than 762 m/s (2,500 fps) is justifiable, as it is replaced by a more conservative approach, which requires a site-specific evaluation when shear wave velocities are less than 762 m/s (2500 fps). Hence, the staff considers Issue 4 of Question 3 in RAI-SRP2.5-RGS1-15 resolved.

The staff also confirmed the changes made in Revision 17 to the Tier 1 and Tier 2 tables to address the issue raised in RAI-SRP2.5-RGS1-15, Question 3, Issue 5 regarding the minimum shear wave velocity. Based on the fact that the applicant revised the criterion for the low-strain shear wave velocity in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, the staff considers Issue 5 of Question 3 in RAI-SRP2.5-RGS1-15 resolved.

The staff also reviewed the applicant's description of the SSE. In Issue 6 of Question 3 of RAI-SRP2.5-RGS1-15, the staff asked the applicant to address the following five concerns related to the SSE: (1) designate the free-field ground motion "CSDRS" instead of "SSE"; (2) review the definition of "outside the range evaluated for the AP1000 design certification" because possible shear-wave velocity inversions were not discussed, but may significantly affect the results of site response and SSI analyses; (3) clarify whether HRHF GRMS were defined at foundation level or in the free field; (4) amend the statement regarding acceptability of site-specific GRMS falling within the AP1000 HRHF to reflect acceptability "over the entire frequency range"; and (5) update DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, to be in agreement with changes made to Section 2s.5. In addition, in RAI-SRP2.5-RGS1-21, Question 3, the staff asked the applicant to further clarify the term HRHF GMRS and the differences between the AP1000 HRHF GMRS and the AP1000 CDRS.

The applicant addressed each item separately in its response. With respect to the staff's first concern, the applicant referred the staff to its response to RAI-SRP2.5-RGS1-02, and stated that "The ground motion response spectra have been revised to the certified seismic design response spectra (CSDRS) instead of the SSE." Since the revised DCD now uses the proper terminology, the staff considers this issue resolved. The applicant addressed the second item in staff's question by referring to its response to RAI-SRP2.5-RGS1-04, and stated that the revised DCD text now states the following: "The shear wave velocity should generally increase with depth. The average low strain shear wave velocity in any layer should not be less than 80 percent of the average shear wave velocity in any layer at higher elevation." Since the applicant clarified the phrase "outside the range evaluated for the AP1000 design certification" as 80 percent of the velocities of the overlying layers, the staff considers this issue resolved. In response to the third concern identified by the staff, the applicant proposed a revision to the DCD and referred the staff to the proposed revisions described in the applicant's responses to RAI-SRP2.5-RGS1-02 and RAI-SRP2.5-RGS1-03. The staff's evaluations of these responses are discussed above. The applicant addressed the fourth staff concern by making a simple revision to include the phrase "over the entire frequency range." Hence, the staff considers this issue resolved. The applicant addressed the fifth item by revising the tables in question and committing to incorporate the revised tables in Revision 17 of the DCD. After reviewing Revision 17 of the AP1000 DCD, the staff determined that the tables in question were revised, but not exactly as specified in the applicant's response to RAI-SRP2.5-RGS1-15. The applicant presented the revision for the site parameter SSE in DCD Tier 1, Table 5.0-1, but not in DCD Tier 2, Table 2-1. Therefore, the staff considered RAI-SRP2.5-RGS1-15 unresolved and tracked this as Open Item OI-SRP2.5-RGS1-15.

To address the staff's concerns described in Open Item OI-SRP2.5-RGS1-15, the applicant submitted a revised response on October 20, 2009, and proposed DCD revisions that are consistent with the commitments made by the applicant in its responses to RAI-SRP2.5-RGS1-02, RAI-SRP2.5-RGS1-03, RAI-SRP2.5-RGS1-04 and RAI-SRP2.5-RGS1-15. In a subsequent revision to the DCD, the applicant incorporated the DCD changes in Tier 1 Table 5.0-1, Tier 2 Table 2-1, and Tier 2 Sections 3.7.1 and 3.7.1.1.

In response to RAI-SRP2.5-RGS1-21, Question 3 the applicant stated that it will replace the term "HRHF GMRS" with "HRHF envelope response spectra" in its next DCD revision. In regard to the issues related to the differences between HRHF and the AP1000 CSDRS, the applicant clarified that the HRHF response spectra are not a second set of CSDRS. The HRHF serves the purpose of determining the acceptability of the site-specific response spectra when there is exceedance in the high-frequency component of the AP1000 CSDRS for a hard rock site. Following further discussions with the staff, the applicant agreed to add sentences to the

Tier 1 table describing when the HRHF frequency could be applied for a site. The added text would state, "Evaluation of a site for application of the HRHF envelope response spectra includes consideration of the limitation on shear wave velocity identified for use of the HRHF envelope response spectra. This limitation is defined by a shear wave velocity at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths."

Since specific shear wave velocities were defined for the soil profile that was used in development of the HRHF envelope response spectra, the applicant stated that it will address the limitation on shear wave velocity in its next DCD revision. The applicant also proposed a DCD revision to reflect the necessary changes. Based on review of the response, the staff finds that: (1) The use of "HRHF envelope response spectra" instead of "HRHF GMRS" will eliminate the confusion between design response spectra and GMRS, because the HRHF response spectra are design basis for hard rock site, while the GMRS is obtained from site-specific seismic response analysis; (2) The applicant stated that the HRHF envelope response spectra are not a second set of design spectra but specifically for hard rock sites with higher seismic response spectra in high frequencies. The applicant also specified the shear wave velocity condition for the hard rock sites where the HRHF envelope response spectra may apply; and (3) the proposed DCD revision will ensure that all necessary changes will be documented in the AP1000 DCD. The staff, therefore, concludes that the response to Question 3 of RAI-SRP2.5-RGS1-21 is adequate. The applicant incorporated conforming changes in a subsequent revision to AP1000 DCD Tier 1, Section 5.0, Tables 5.0-1, 5.0-3 and 5.0-4; Tier 2, Table 2-1; Sections 2.5.2, 3I.1, and 3I.2; and Figures 3I.1-1 and 3I.1-2. Therefore, this issue is resolved.

2.5.2.4.2 Sites with Geoscience Parameters outside the Certified Design

In Section 2.5.2.3, the applicant stated that, if soil conditions are outside the range evaluated for the AP1000 DC, a site-specific evaluation can be performed. The staff asked the applicant, in RAI-SRP2.5-RGS1-05, to provide acceptance criteria regarding soil properties. In RAI-SRP2.5-RGS1-06, the staff asked the applicant to state the requirements for a site-specific soil degradation model that is one of the basic inputs to the SSI analysis in the AP1000 DCD. In response to these questions, the applicant indicated that: (1) it would add the requirement for a site-specific soil degradation model in a later revision of the DCD; and (2) Section 3.7.1.4 of the DCD provides tables and figures illustrating soil properties that were used for the design of the nuclear island. The applicant stated that COL applicants referencing the AP1000 DCD would generate site-specific soil profile plots and compare them with the design presented in Section 3.7.1.4. The applicant also stated that it revised DCD Table 3.7.1.4 to reflect the strain compatible properties. The staff considers RAI-SRP2.5-RGS1-06 resolved as the applicant implemented the staff's recommendation and revised the DCD to explicitly state that site-specific soil degradation models are a part of the site-specific soil conditions. Since the applicant stated in its response to RAI-SRP2.5-RGS1-05 that Section 3.7.1.4 of the DCD provides tables and plots that can be used by a COL applicant to compare the site soil profile to determine if the soil conditions are outside the range evaluated for the AP1000 DC, the staff concludes that the applicant's response provided an adequate description of how a COL applicant would assess whether the soil conditions at a site are outside the range defined by the DCD and considers question RAI-SRP2.5-RGS1-05 resolved. Based on the RAI responses from the applicant and review of Section 3.7.1.4, the staff concludes that the applicant provided adequate information to resolve RAI-SRP2.5-RGS1-05 and RAI-SRP2.5-RGS1-06.

The staff considered the incorporation of APP-GW-CLE-004 into DCD Section 2.5.2.3. In RAI-SRP2.5-RGS1-16, the staff asked the applicant to define the term "geoscience parameters" used in the subtitle of Section 2.5.2.3, "Sites with Geoscience Parameters Outside the Certified Design." In addition, the staff also asked the applicant to clarify the discrepancy between DCD Section 2.5.2.3 and DCD Section 3.7.1.1. DCD Section 2.5.2.3 states that a site-specific evaluation can be performed if the site-specific spectra at foundation level exceed the response spectra at any frequency or if the soil conditions are outside the range evaluated in Section 2.5.2.3. DCD Section 3.7.1.1 states that design response spectra are applied at the foundation level in the free field at hard rock sites and at finished grade in the free field at firm rock and soil sites. The staff also asked the applicant to clarify the statement that the site design response spectra at the foundation level in the free-field were used to develop the floor response spectra, which is inconsistent with DCD Section 3.7.1.1 for soil sites.

In its response, the applicant stated that DCD Section 2.5.2.3 was re-written based on the staff's question RAI-SRP2.5-RGS1-04 and referred the staff to its response to RAI-SRP2.5-RGS1-04. In that response, the applicant stated that it revised the title of Section 2.5.2.3 from "Sites with Geoscience Parameters Outside the Certified Design" to "Site Specific Seismic Evaluation." With this revision, the staff considers the first issue closed since the applicant revised the title and eliminated the questioned phrase. The applicant also clarified the apparent discrepancy between DCD Section 2.5.2.3 and Section 3.7.1.1 by revising its response to RAI-SRP2.5-RGS1-04. The applicant revised the DCD to state that "If the site-specific spectra at foundation level at a hard rock site or at grade for other sites exceed the certified seismic design response spectra in Figures 3.7.1-1 and 3.7.1-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed." With this revision, the staff concludes that the apparent discrepancy has been eliminated and the issue resolved. The applicant also modified the DCD to clarify the statement outlined above by the staff's question. The revised DCD clarified this third issue. The DCD text now reads "The certified design response spectra in the free-field given in Figures 3.7.1-1 and 3.7.1-2 were used to develop the floor response spectra." With this revision, the staff considers the third issue in the staff's question above resolved.

2.5.2.5 Post Combined License Activities

The staff will identify post-COL activities on a site-by-site basis as part of the review of a COL application referencing the AP1000 DCD.

2.5.2.6 Conclusions

Based on the review of Revision 17 of the AP1000 DCD Tier 2, Section 2.5.2; Tier 1, Table 5.0-1 (and Tier 2, Table 2-1); and APP-GW-GLE-004, the staff finds that the applicant adequately detailed how to determine site-specific GMRS, specified criteria for a site to be suitable for the AP1000 standard design, and provided detailed guidance on performing site-specific seismic evaluation for sites that do not meet the scope of the seven siting requirements described in the DCD. The applicant also provided a set of site parameters related to the geological and seismological basis for the AP1000 standard design, such as requirements on SSE and associated site response spectra, fault displacement potential, and the subsurface material lateral variability requirement. The staff concludes that the geological and seismological related site parameters and requirements presented in the DCD are acceptable and meet the regulatory requirements of 10 CFR 100.23, GDC 2, and 10 CFR 52.47(a)(1).

The applicant submitted changes to the DCD that provide the seismic design and supporting analysis for a range of soil conditions representative of expected applicants for a COL referencing the AP1000 design. These changes provide increased standardization for this aspect of the design. In addition, these changes reduce the need for COL applicants to seek departures from the current AP1000 design, since many sites do not conform to the currently-approved hard rock sites. Therefore, the change increases standardization and meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

2.5.3 Surface Faulting

The applicant changed the site parameter provided in Tier 1, Table 5.0-1 and Tier 2, Table 2-1, for "Fault Displacement Potential" from "None" in Revision 15 to "Negligible" in Revision 17 of AP1000 DCD. The staff, in Question 1 of the RAI-SRP2.5-RGS1-21, asked the applicant to clarify the definition of "negligible." In its response to this question, the applicant first explained that the reason of making this change is because of the difficulty for a COL applicant to demonstrate that the fault displacement potential for a site is absolutely "None." Following further discussions with the staff, the applicant subsequently proposed to change this site parameter to "No potential fault displacement considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures." The staff considers that no fault displacement potential beneath these structures is a reasonable design basis for representing most existing nuclear power plant sites, as well as the ESP and COL application site. DCD Section 2.5.3 describes the information on surface faulting that the COL applicant needs to provide to satisfy the requirement for no surface faulting by completing geological, seismological, and geophysical investigations. Therefore, the staff concludes that this design site parameter is acceptable because it is consistent with the guidance in RG 1.206, and can reasonably ensure that the regulatory requirements of 10 CFR 100.23 will be met. Accordingly, the issue of clearly defining the site parameter for fault displacement potential in Question 1 of the RAI-SRP2.5-RGS1-21 is resolved. The applicant incorporated conforming changes in a revised subsequent revision to AP1000 DCD Tier 1, Table 5.0-1 and Tier 2, Table 2-1.

2.5.4 Stability and Uniformity of Subsurface Materials and Foundations

2.5.4.1 Introduction

Section 2.5.4, "Stability and Uniformity of Subsurface Materials and Foundations," of the AP1000 DCD presents the requirements related to stability of subsurface materials and foundations for COL applicants referencing the AP1000 standard design. The site-specific information includes excavation, bearing capacity, settlement, and liquefaction potential.

2.5.4.2 Technical Information in the Application

2.5.4.2.1 Excavation

Section 2.5.4.1 of the AP1000 DCD provides the requirements for site excavation. In this section, the applicant stated that, for the nuclear island structures below grade, a COL applicant may use either a sloping excavation or a vertical face. The applicant further stated that, if a COL applicant uses a sloping excavation, an evaluation of the 3-D effects on the site response and site-specific SSI analyses must be performed using a combination of either 2-D or 3-D SASSI models that reflect the sloping excavations. In the event that a vertical face is used, the

COL applicant would need to cover the face with a waterproof membrane, as described in DCD Section 3.4.1.1.1.1, or use soil nailing and mechanically stabilized earth (MSE) walls as the outside form for the exterior walls below grade of the nuclear island.

DCD Section 2.5.4.1.1 describes the detailed requirements for using a soil nailing method as an alternative to stabilize vertical faces of undisturbed soil or rock below grade for nuclear island structures. The applicant stated that the soil nailing method produced a vertical surface down to the bottom of the excavation and was used as the outside form for the exterior walls below grade of the nuclear island. The applicant also provided details on soil-retaining wall installation in this section.

DCD Section 2.5.4.1.2 describes the MSE as a flexible retaining wall using strip, grid, or sheet type of tensile reinforcements so that the wall behaves as a retaining wall. The applicant stated that the tensile strength of the reinforcements provides internal stability and the walls could be used in areas where retaining wall soils have been removed or elevation needs to be raised.

DCD Section 2.5.4.1.3 describes the mud mat, including both the upper and lower mats, which will be placed ahead of the placement of reinforcements for the foundation mat structural concrete. The applicant stated that both the lower mud mats would have a compressive strength of 17,236 kPa (2,500 psi) and be a minimum of 15.24 cm (6 in) thick. Finally, DCD Section 3.4.1.1.1.1 describes waterproofing system alternatives.

2.5.4.2.2 Bearing Capacity

DCD Section 2.5.4.2, “Bearing Capacity,” specifies that the maximum bearing reaction is less than 1,676 kPa (35,000 pounds per square foot (psf)) under all combined loads, including the SSE, based on the analyses described in Appendix 3G to the AP1000 DCD and occurs at the western edge of the shield building. The DCD applicant noted that the COL applicant would need to verify whether the site-specific allowable soil-bearing capacities for static and dynamic loads would exceed this demand with a factor of safety appropriate for the design load combination, including SSE loads.

In DCD Tier 1, Table 5.0-1, and Tier 2, Table 2-1, the applicant listed the site parameters of average allowable bearing capacity. These tables stated the average allowable static soil bearing capacity as greater than or equal to the average bearing demand of 8,900 psf over the footprint of the nuclear island at its excavation depth. It also defined the maximum allowable dynamic bearing capacity for normal plus SSE loads as greater than or equal to the maximum bearing demand of 35,000 psf at the edge of the nuclear island at its excavation depth, or by performing site-specific analyses to demonstrate a factor of safety appropriate for normal plus SSE loads.

2.5.4.2.3 Settlement

DCD Section 2.5.4.3, “Settlement,” requires the COL applicant to address both short-term (elastic) and long-term (heave and consolidation) settlement for soil sites for the history of loads imposed on the foundation consistent with the construction sequence. The applicant noted that the time-history of settlements should include construction activities and construction of the superstructure. The applicant also stated that the AP1000 design does not rely on SSCs located outside the nuclear island footprint for safety-related functions.

In Revision 17 of the AP1000 DCD, the applicant added Table 2.5-1 which provides guidance to the COL applicant on predictions of absolute and differential settlement that are acceptable without additional evaluation.

2.5.4.2.4 Liquefaction

In DCD Section 2.5.4.4, the DCD applicant stated that the COL applicant will demonstrate that, for soil sites, the potential for liquefaction is negligible for both the soil underneath the nuclear island foundation and at the side embedment engaged in passive resistance adjacent to the nuclear island. DCD Tier 1, Table 5.0-1, as well as Tier 2, Table 2-1, state that liquefaction potential is negligible at the site.

2.5.4.2.5 Subsurface Uniformity

Section 2.5.4.5 of the DCD states that, although the design and analysis of the AP1000 was based on soil or rock conditions with uniform properties within horizontal layers, provisions and design margins to accommodate many nonuniform sites were also included. The applicant described, in detail, the types of site investigation that would be sufficient for a “uniform” site or a “nonuniform” site. The applicant indicated that the acceptability of a nonuniform site would be based on an individual site evaluation. The applicant concluded that, for uniform sites whose site parameters fall within the site profiles evaluated as part of the DC, no further action will be needed. However, for nonuniform sites, or other sites whose parameters do not fall within the site profiles, a site-specific evaluation will need to be performed. For nonuniform sites, Sections 2.5.1 and 2.5.4.6.1 of the DCD outline the geological investigations for the extended investigation effort to determine whether the site is acceptable for construction of an AP1000 reactor. In Revision 17 of the DCD, the applicant deleted Sections 2.5.4.5.1 and 2.5.4.5.2 and labeled them as “Not Used.”

2.5.4.2.5.1 Site Foundation Material Evaluation Criteria

DCD Section 2.5.4.5.3 states that the COL applicant will demonstrate that the variation of subgrade modulus across the nuclear island footprint will be within the range considered for design of the nuclear island basemat. The DCD also stated that the COL applicant will consider the subsurface conditions within the nuclear island footprint and 12.2 m (40 ft) beyond, and to a depth of 36.6 m (120 ft) below finished grade within the nuclear island footprint. The applicant also noted that a uniform site would be acceptable for the AP1000 design, without additional site-specific analyses, based on the analyses and evaluations performed to support the DC. The applicant also outlined two criteria for site uniformity.

2.5.4.2.5.2 Site-Specific Subsurface Uniformity Design Basis

DCD Section 2.5.4.5.3.1 states that nonuniform soil conditions may require the evaluation of the AP1000 seismic response, as described in DCD Section 2.5.2.3.

For the rigid basemat evaluation, the applicant stated that if the site variability can be identified without significant variations in the horizontal direction, a 2-D analysis can be used. However, the applicant also stated that sites with variability in the horizontal direction indicate the need for a 3-D analysis. The applicant further stated that the bearing pressure from the site-specific analysis needs to be less than or equal to 120 percent of that for a similar site with uniform soil properties.

For a flexible basemat evaluation, the applicant stated that soils may be represented by soil springs or by a finite element model, depending on the variability identified at the site. The applicant also pointed out that, for a site to be acceptable, the bearing pressures from the site-specific analyses will need to be less than the design bearing strength of each portion of the basemat under both static and dynamic loads.

In DCD Tier 1, Table 5.0-1, the applicant addressed the site parameters for lateral variations by stating that the soils supporting the nuclear island should not have extreme variations in subgrade stiffness. The applicant described the documentation of variations as follows:

- Soils supporting the nuclear island are uniform in accordance with RG 1.132 if the geologic and stratigraphic features at depths less than 36.6 m (120 ft) below grade can be correlated from one boring or sounding location to the next with relatively smooth variations in thicknesses or properties of the geologic units; or

- Site-specific assessment of subsurface conditions demonstrates that the bearing pressures below the footprint of the nuclear island do not exceed 120 percent of those from the generic analyses of the nuclear island at a uniform site; or

- Site-specific analysis of the nuclear island basemat demonstrates that the site-specific demand is within the capacity of the basemat.

The applicant further stated that, as an example of sites that are considered uniform, the variation of shear wave velocity in the material below the foundation to a depth of 36.6 m (120 ft) below finished grade within the nuclear island footprint and 12.2 m (40 ft) beyond the boundaries of the nuclear island footprint meets the criteria in the case outlined below.

Case 1: For a layer with a low-strain shear wave velocity greater than or equal to 2,500 fps, the layer should have approximately uniform thickness, should have a dip not greater than 20 degrees, and should have less than 20-percent variation in the shear wave velocity from the average velocity in any layer.

DCD Tier 1, Table 5.0-1, also states that the shear wave velocity should be greater than or equal to 305 m/s (1,000 ft/s) based on minimum low-strain soil properties over the footprint of the nuclear island at its excavation depth.

2.5.4.2.6 Combined License Information

In response to RAI-TR-85-SEB1-36 (Revision 4 dated October 22, 2010), the applicant proposed to revise Section 2.5.4.6.11 to state that the COL applicant will provide data on short-term (elastic) and long-term (heave and consolidation) settlement for soil sites for the history of loads imposed on the nuclear island foundation and adjacent buildings consistent with the construction sequence. The response also specifies that special construction requirements will be described, if required, to accommodate settlement predicted to exceed the design settlement limits.

In response to RAI-TR-85-SEB1-17 (Revision 5 dated July 15, 2010), the applicant proposed to revise Section 2.5.4.6.11 to state that Section 3.8.5.4.2 includes analyses of settlement during construction completed to support the DC and the required limitations on construction sequence for some sites. The limitations on construction sequence impose limits on the placement of concrete for the shield building and the auxiliary building prior to completion of both buildings at elevation 25.15 m (82.5 ft).

In response to RAI-TR-85-SEB1-35 (Revision 3 dated, June 30, 2010), the applicant proposed to add Section 2.5.4.6.12, “Waterproofing System” to the DCD. This section states that the COL applicant shall provide a waterproofing system used for the foundation mat (mudmat) and below grade exterior walls exposed to flood and groundwater under seismic Category I structures. It specifies that the waterproofing membrane should be placed immediately beneath the upper mudmat and on top of the lower mudmat. This section also refers the detailed performance requirements for the waterproofing system to Section 3.4.1.1.1.1.

All COL information items are summarized in AP1000 DCD Tier 2, Table 1.8-2, “Summary of AP1000 Standard Plant Combined License Information Items.”

2.5.4.3 Regulatory Basis

The applicable regulatory requirements and guidance for reviewing the applicant’s discussion of stability of subsurface materials and foundations are as follows:

- 10 CFR Part 50, Appendix A, GDC 2, as it relates to consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

- 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” as it applies to the ability of the design of nuclear power plant SSCs important to safety to withstand the effects of earthquakes.

- 10 CFR 100.23, which provides the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants.

- RG 1.132

- RG 1.138, “Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants”

- RG 1.206

2.5.4.4 Evaluation

2.5.4.4.1 Excavation

In DCD Section 2.5.4.1, the applicant stated that if a sloping excavation was used for a site, then the 3-D effect on the SSI analysis should be considered. In RAI-SRP2.5-RGS1-07, the staff asked the applicant to add this statement to the DCD as a requirement for COL applicants. In response to this RAI, the applicant added a requirement for the COL applicant to evaluate the 3-D effects by performing a site-specific SSI analysis using either 2-D or 3-D SASSI models, or both, for sloping excavations. The staff reviewed DCD Revision 17 and confirmed that the applicant had included this updated information. Accordingly, the staff considers the revised DCD to be sufficient to resolve RAI-SRP2.5-RGS1-07, which requested that the applicant include the requirement to evaluate the 3-D effects through site-specific SSI analyses in the DCD.

Since the staff found that at least one COL applicant used precast facing panels to retain the side soil, RAI-SRP2.5-RGS1-08 asked the applicant to clarify whether it would revise the DCD regarding other methods that can be used to retain the vertical excavation face. In response to this RAI, as well as to RAI-TR85-SEB1-040, the applicant stated that it substantially revised Section 2.5.4.1 to address the option of using an MSE wall with precast concrete facing panels to retain the side soil. The staff reviewed the revisions to the DCD, particularly the option to use an MSE wall, and concludes that the additional options to retain side soil are sufficient to resolve the geotechnical engineering aspects of RAI-SRP2.5-RGS1-08. Therefore, the staff considers this RAI resolved.

2.5.4.4.2 Bearing Capacity

Based on its review of Section 2.5.4.2, the staff raised the following concerns in RAI-SRP2.5-RGS1-09:

- Since bearing capacity is highly site-specific, replace the “bearing capacity” value calculated from seismic analyses with the “bearing demand” value based on the maximum foundation contact pressure.

- Justify why Revision 16 states that the maximum allowable dynamic bearing capacity (bearing demand) is greater than or equal to 1,676 kPa (35,000 psf), which is far less than 5,746 kPa (120,000 psf), as listed in the prior revision of DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1.

- Define the “factor of safety” for the bearing capacity evaluation.

In response to this RAI, the applicant replaced the term “bearing capacity” with “bearing demand” in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, and changed average allowable static soil bearing capacity from 421 kPa (8,600 psf) to 426 kPa (8,900 psf) to reflect the enhanced shield building design. Revision 17 of the DCD includes these changes, and the staff considers Issue 1 of RAI-SRP2.5-RGS1-09 resolved.

In response to Issue 2 above, the applicant referred the staff to its response to RAI-TR85-SEB1-03 for an explanation as to why Revision 16 of the AP1000 DCD listed the bearing capacity value of 1,676 kPa (35,000 psf). In responding to the RAI, the applicant stated that this difference resulted from: (1) Different seismic loads being applied to the foundation dynamic response analysis. The prior revision used a seismic load for hard rock certified design, while the current version used a design that envelops all rock and soil cases; and (2) the prior revision used the results from a more conservative equivalent static analyses, while the current version used the result from a nonlinear dynamic analyses. The dynamic nonlinear analyses showed a much lower bearing reaction (1,331 kPa (27,008 psf) for hard rock) than those from the equivalent static design analyses for the basemat. Using the commercial computer software 2-D ANSYS, the applicant completed nonlinear analyses, which yielded higher bearing pressures (1,652 kPa (34,500 psf)) for a SM soil case than those for the hard rock case. Based on the new analysis results, the applicant chose the soil bearing reaction of 1,676 kPa (35,000 psf) to cover both soil and rock sites. The applicant further indicated that the bearing pressures from the ANSYS analyses were conservative because the effect of the side soil was neglected. Since the applicant re-analyzed the bearing capacity calculations using a more realistic non-linear soil model, the staff considers this reduction in bearing capacity value as acceptable since the non-linear model would result in more realistic estimates than the previous equivalent static analyses the applicant conducted. Hence, the staff considers Issue 2 resolved.

Regarding the factor of safety used for the bearing capacity evaluation, the applicant stated that the factor of safety should be site-specific and, therefore, COL applicants will be responsible for defining an appropriate factor of safety for their sites. Since this issue will be addressed by each COL applicant, the staff considers Issue 3 in RAI-SRP2.5-RGS1-09 resolved.

After reviewing the applicant's response, including the revision of the DCD, the explanation of the allowable bearing capacity, and the site-specific nature of the factor of safety, the staff concludes that the applicant provided adequate information to address all three areas of concern identified in RAI-SRP 2.5-RGS1-09. However, since RAI-SRP2.5-RGS1-09 also relates to another RAI related to structural engineering (RAI-TR85-SEB1-03), the staff will not consider the RAI resolved until the applicant adequately addresses the structural engineering concerns. This issue was tracked as Open Item OI-SRP2.5-RGS1-09.

To close Open Item OI-SRP2.5-RGS1-09, the applicant provided a response to RAI-TR85-SEB1-03, dated September 18, 2007. In the response, the applicant provided detailed explanations of the soil model used in the 3-D ANSYS finite element model and how it determined the maximum dynamic bearing pressure. In a later response dated October 20, 2009, the applicant also provided a new maximum bearing demand value that is based on a 3-D SASSI analyses. As a result of these new analyses, a more realistic and conservative limit of maximum bearing seismic demand will now be used as a site parameter in the DCD. Based on the review of the applicant's responses to RAI-TR85-SEB1-03 and Open Item OI-SRP2.5-RGS1-09, the staff concludes that the analysis model used in the dynamic bearing pressure determination is adequate and that the design parameter specified in the DCD is reasonable. Because the applicant adequately addressed all issues identified in RAI-TR85-SEB1-03, Open Item OI-SRP2.5-RGS1-09, and RAI-TR85-SEB1-03, and also because the staff confirmed that the applicant revised related site parameters in AP1000 DCD, Open Item OI-SRP2.5-RGS1-09 is closed.

While reviewing this section, the staff also considered the information provided in APP-GW-GLE-004 and DCD Tier 1, Table 5.0-1. The staff asked the applicant, in Questions 1 and 2 of RAI-SRP2.5-RGS1-15, to clarify the use of the terms, "average allowable static soil bearing capacity," and "average allowable dynamic soil bearing capacity," and justify the use of the phrase "greater than or equal to" for the calculated soil bearing demand values. In its response, the applicant cited the proposed changes to DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, made in response to RAI-SRP2.5-RGS1-09, which include the definitions of average allowable static and dynamic bearing capacity. In response to the second question, the applicant stated that site-specific allowable bearing capacity must be "greater than or equal to" the AP1000 calculated demand values. Since the staff had already determined that the revisions to the two tables were acceptable in RAI-SRP2.5-RGS1-09, the staff concludes that Question 1 of RAI-SRP2.5-RGS1-15 is resolved. Furthermore, the staff considered the statement of requiring the site-specific allowable bearing capacity to be greater than or equal to the calculated demand values and concludes that this statement sufficiently addresses the geotechnical engineering concerns of the second question of RAI-SRP2.5-RGS1-15. Accordingly, the staff considers Questions 1 and 2 of RAI-SRP2.5-RGS1-15 to be resolved.

In RAI-SRP2.5-RGS1-21, Question 2, the staff also requested the applicant redefine the site parameter for dynamic bearing capacity, which is labeled as “Maximum Allowable Dynamic Bearing Capacity for Normal Plus Safe Shutdown Earthquake (SSE)” used in AP1000 DCD, Revision 17, Tier 1, Table 5.0-1 and Tier 2, Table 2-1. The staff considered this label to not clearly define the requirement that a site must have the minimum capacity to meet the maximum dynamic bearing demand. Therefore, in Question 2 of RAI-SRP2.5-RGS1-21, the staff asked the applicant to justify the use of “Maximum Allowable” for dynamic bearing capacity parameter. In response to this RAI, the applicant stated that the modifier “maximum allowable” was not necessary and proposed to eliminate it from the referenced tables. Based on this proposed change the staff considers Question 2 of RAI-SRP2.5-RGS1-21 resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

2.5.4.4.3 Settlement

In response to RAI-TR85-SEB1-36 (Revision 4 dated October 22, 2010), regarding the settlement criteria, the applicant proposed the following revisions to the AP1000 DCD:

1. Adding “Limits Of Acceptable Settlement Without Additional Evaluation” site parameter in Tier 1, Table 5.0-1, which specifies the design settlement limits.

2. Adding detailed settlement criteria in DCD Section 2.5.4.3 to specify that the predicted settlements will cover the periods before and through the construction phase, and for the subsequent plant operating period or otherwise justified. The COL applicant needs to provide detailed evaluation and construction sequence plan if the predicted settlements exceed the limits of design settlements. For a soil site, settlements would be measured and compared to the predicted settlement values during construction and plant operation, and any exceedances would require additional investigation.

3. Procedures for additional settlement evaluation were provided, the word “suggested” being removed from the characterization of the alternatives. The procedures include evaluating the impact of the elevated estimated settlement values on the critical components of the AP1000 structures; submitting a construction sequence to control the predicted settlement behavior; providing a uniform excavation and engineered backfill to manage static building rotation and differential settlement between the nuclear island and adjacent structures; and implementing an active settlement monitoring system throughout the entire construction sequence and plant operation (a long-term plan). The proposed DCD revision also specifies primary elements in the settlement monitoring system, and requires that the settlement data to be maintained during construction and post-construction, as needed, depending on the field measurement results.

The staff reviewed the settlement requirements for the AP1000 reactor, as specified in Tier 1, Table 5.0-1 and Tier 2, Section 2.5.4.3 and Table 2.5-1 of the AP1000 DCD, and the assertion that because of the locations of all safety-related structures on the nuclear island, the differential settlement requirements are defined for adjacent structures. The staff also reviewed the proposed methods for additional evaluation if the predicted settlements exceed the design limits at a COL site, and the requirement for implementing an active settlement monitoring system throughout the entire construction sequence, including plant operation (long-term plan), for a soil site. The staff concludes that the applicant adequately described settlement criteria and provided clear requirements and detailed evaluation procedure for COL applicants referencing the AP1000 DCD to follow. Therefore, the settlement requirements described in the DCD are sufficient and acceptable. However, the applicant proposed to revise the settlement requirements in Tier 2, Section 2.5.4.3, “Settlement,” and to add the settlement site parameter to Tier 1, Table 5.0-1, as stated in its response to RAI-TR-85-SEB1-36. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

2.5.4.4.4 Liquefaction

During the review of DCD Section 2.5.4.4, the staff noted that DCD Tier 1, Table 5.0-1 and DCD Tier 2, Table 2-1 in Revision 15, stated the liquefaction potential at the plant site as “NONE.” In Revision 16 of DCD Section 2.5.4.4, the applicant changed ‘NONE’ to “NEGLIGIBLE.” In separate questions, RAI-SRP2.5-RGS1-11 and Question 1 in RAI-SRP2.5-RGS-21, the staff asked the applicant to define how and where the potential for liquefaction was negligible at a site. In response to these RAIs, the applicant revised Section 2.5.4.4 to define that, for a soil site, the COL applicant should demonstrate that the potential for liquefaction was negligible for both the soil underneath the nuclear island foundation and the soil of the side embedment engaged in passive resistance adjacent to the nuclear island. The applicant restated in DCD Revision 17 that for the AP1000 liquefaction beneath the certified design. The applicant further stated:

> The AP1000 design has not been evaluated for a site where there is a liquefaction potential of the soil below the nuclear island. A COL applicant must describe the soil and rock structure beneath the nuclear island in their application. DCD Subsection 2.5.4.6 describes the geotechnical information that should be provided by the COL applicant. Liquefaction potential for the site is evaluated for the site specific SSE ground motion (specific site GMRS). A COL applicant will satisfy the requirement for no liquefaction by providing information concerning the properties and stability of supporting soils and rock consistent with the guidance of regulatory guide 1.206.

Regarding the word change from “None” to “Negligible,” the applicant explained that the reason for making this change is because of the difficulty for a COL applicant to demonstrate the liquefaction potential at a site as absolutely “None.” The applicant, however, recognized that the AP1000 design has not been evaluated for a site where there is a liquefaction potential of the soil below the nuclear island. Following further discussions with the staff, the applicant subsequently proposed to change this site parameter to “No liquefaction considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.” The staff considers that no potential liquefaction beneath these structures at a site is a reasonable design basis for representing most of the existing nuclear power plant sites, as well as ESP and COL application sites. DCD Section 2.5.4.6 describes the information concerning the properties and stability of supporting soils and rock that the COL applicant needs to provide in order to evaluate the liquefaction potential beneath the nuclear island and to satisfy the requirement of no liquefaction potential. Therefore, the staff concludes that this design site parameter is acceptable because it is consistent with the guidance of RG 1.206, and can reasonably ensure the regulatory requirements of 10 CFR 100.23 are met.

Based on the applicant’s responses and the staff’s confirmation that Revision 17 of the AP1000 DCD includes these revisions, the staff concludes that the applicant clarified the liquefaction potential requirement and sufficiently addressed the concerns of the RAIs. Accordingly, the staff considers RAI-SRP2.5-RGS1-11 and Question 1 of RAI-SRP2.5-RGS-21 resolved provided the proposed changes are incorporated in the revised DCD Tier 1 Table 5.0-1 and Tier 2, Table 2-1. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

2.5.4.4.5 Subsurface Uniformity

At the end of Section 2.5.4.5, Revision 15 of the DCD presented a survey of 22 commercial nuclear power plant sites in the United States that focused on site parameters that affect the seismic response. All but one of the 22 sites were uniform sites. In RAI-SRP2.5-RGS1-12, the staff questioned the purpose of this survey and the reasons for its inclusion in the AP1000 DCD. As a response to this RAI, the applicant removed the paragraph referencing the survey, having decided that it was no longer applicable. Since the questionable paragraph has been removed, the staff considers RAI-SRP2.5-RGS1-12 to be resolved.

Regarding the site investigation criteria, in RAI-SRP2.5-RGS1-13, the staff asked the applicant to explain why it addressed issues related to settlement caused by static loads but did not consider the criteria needed to evaluate site response and dynamic SSI issues. In response to this RAI, the applicant revised the DCD to remove Sections 2.5.4.5.1 and 2.5.4.5.2, stating that the site investigation criteria should not be part of the DCD, but should be part of the COL applicant’s submittal. Since the content in question was removed from the DCD, the staff considers this RAI resolved.

In RAI-SRP2.5-RGS1-14, the staff asked the applicant to clarify and provide the basis for evaluation criteria for the site uniformity discussed in APP-GW-GLE-004. The applicant responded by referring to the evaluation criteria given in DCD Section 2.5.4.5, as revised in the technical report. The applicant stated that the AP1000 would be acceptable at uniform sites without further evaluation based on the definition of uniform given in RG 1.132. The applicant justified the acceptability of relatively smooth variations by citing design analyses of the basemat described in DCD Section 3.8.5, which considered the basemat to be supported by uniform soil springs. Furthermore, the applicant indicated that the AP1000 design included a 20-percent margin above the results of uniform soil springs to accommodate the smooth variations that may occur at a uniform site. Finally, the applicant stated that, although additional evaluation would be required for nonuniform sites, the level of detail would depend on the nonuniformity identified in the site investigations.

The staff considered this response, particularly the 20-percent margin above uniformity of soil springs, as well as the applicant’s adoption of the definition of “uniform” as described in RG 1.132, and concluded that the applicant adequately addressed the concern of variations in uniformity of the site identified in the RAI. Therefore, the staff considers RAI-SRP2.5-RGS1-14 resolved.

In Question 3 of RAI-SRP2.5-RGS1-15, the staff asked the applicant to: (1) clarify the definition of uniform soils in Criterion 1 and address the incorporation of specific criteria on shear wave and compressional wave velocity profiles needed to ensure the adequacy of SSI calculations; (2) clarify how the variability in bearing pressure relates to the corresponding variability of the soil stiffness and shear wave velocity and describe the basis of Criterion 2; and (3) provide the basis for using the phrase “within the NI [nuclear island] footprint” in describing Criterion 3, since the zone of influence under the foundation level would extend beyond the boundary of the nuclear island foundation mat.

The applicant responded to the first issue of Question 3 by stating that, while the uniformity conditions of RG 1.132 were subjective, for sites where uniformity was not clear, the site will be evaluated as nonuniform. The applicant provided more discussion on shear wave velocity profiles in DCD Section 2.5.2. With respect to the second issue, the applicant stated that the AP1000 design included a 20-percent margin above the results of the uniform soil springs analyses to accommodate relatively smooth variation in soil springs at uniform sites. The applicant further stated that the member forces and required reinforcement were conservatively assumed to increase in the same percentage as bearing pressure. With respect to the third issue of Question 3, the applicant reiterated information from Paragraph 3 of DCD Section 2.5.4.5.3 stating that it will add the phrase "and 40 feet [12.2 m] beyond the boundaries of the nuclear island footprint" to both DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1.

The staff reviewed the applicant's response and confirmed that the applicant updated DCD Tier 1, Table 5.0-1, and DCD Tier 2, Table 2-1, in Revision 17 of the DCD with the additions described in the RAI response. The staff concludes that the applicant provided sufficient information to address the concerns of site uniformity, uniform soil springs analyses, and the zone of influence at the nuclear island foundation mat. Accordingly, the staff considers Issues 1 through 3 of Question 3 of RAI-SRP2.5-RGS1-15 resolved.

In RAI-SRP2.5-RGS1-17, the staff asked the applicant to explain the applicability of the survey of nuclear power plant conditions in the United States and how the survey results can be used to justify the site uniformity of a prospective site. In response to this RAI, the applicant pointed out that it had deleted the paragraph regarding the survey of nuclear plant conditions in response to RAI-SRP2.5-RGS1-12. Since RAI-SRP2.5-RGS1-12 is already considered resolved, the staff concludes that RAI-SRP2.5-RGS1-17 is also resolved.

In RAI-SRP2.5-RGS1-18, the staff asked the applicant to incorporate in DCD Section 2.5.4.5.1, the potential effects of a lack of uniformity outside the nuclear island footprint in SSI responses. In response to this RAI, the applicant referred to its response to RAI-SRP2.5-RGS1-13, in which the applicant stated that it planned to delete DCD Sections 2.5.4.5.1 and 2.5.4.5.2. Since RAI-SRP2.5-RGS1-13 is resolved, the staff concludes that RAI-SRP2.5-RGS1-18 is also resolved.

In RAI-SRP2.5-RGS1-19, the staff asked the applicant to clarify why it did not discuss faulting criteria. The applicant responded that, although faulting was not discussed as a separate criterion, faulting may result in different soil properties on each side of a fault and that, therefore, the difference in properties would be evaluated against the criteria for lateral variability. The staff reviewed this response and finds that an assessment of lateral variability of soils will be an acceptable substitute to faulting criteria because it will address the offset of the fault in the site area. Therefore, the staff concludes that RAI-SRP2.5-RGS1-19 is resolved.

Finally, in RAI-SRP2.5-RGS1-20, the staff asked the applicant to justify the exclusion of site uniformity evaluation criteria for the case of a soil layer with a low-strain shear wave velocity less than 762 m/s (2,500 fps). In its response, the applicant referred to RAI-SRP2.5-RGS1-15 Question 3, Issue 4, which stated that soil sites would require a site-specific evaluation because of the unrealistically tight limit of ±10 percent. The staff resolved this question in its review of the applicant's response to RAI-SRP2.5-RGS1-15. Therefore, the staff concludes that RAI-SRP2.5-RGS1-20 is resolved.

2.5.4.4.6 Combined License Information

In AP1000 DCD Tier 2 DCD Section 2.5.4.6, the applicant summarizes all COL information items related to geotechnical engineering aspects of a site, with brief descriptions and pointers to related DCD sections, that COL applicants referencing the AP1000 design must address. The COL information items are also listed in AP1000 DCD Tier 2, Table 1.8-2. The staff reviewed this section and concluded that it is necessary to summarize all COL information items to ensure that COL applicants adequately address those items in the COL application to meet the design requirements; therefore, this section is acceptable. However, since the applicant proposed to revise Section 2.5.4.6.11, "Settlement of Nuclear Island," to add Section 2.5.4.6.12, "Waterproofing System," to DCD Tier 2, Section 2.5.4.6; and to add COL Information Item 2.5-17 to DCD Tier 2, Table 1.8-2, as stated in its response to RAI-TR-85-SEB1-17, RAI-TR-85-SEB1-35 and RAI-TR-85-SEB1-36. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

2.5.4.5 Post Combined License Activities

The staff will identify post-COL activities on a site-by-site basis as part of its review of a COL application referencing the AP1000 DCD.

2.5.4.6 Conclusions

Based on its review of Revision 17 of AP1000 DCD Section 2.5.4; DCD Tier 1, Table 5.0-1, and Tier 2, Table 2-1; and APP-GW-GLE-004, as well as the applicant's responses to RAIs and open items, the staff finds the following:

- The applicant described the requirements for site excavation and backfill used for safety-related structure foundations, as well as the requirement for soil retaining structures for COL applicants that reference the AP1000 standard design. The staff finds this acceptable.

- The applicant presented the technical basis for establishing proper static and dynamic foundation bearing capacity requirements, which consider the design static and dynamic loadings, including SSE seismic loading. The staff finds this acceptable.

- Based on the previous review and evaluation performed by the staff, as well as the proposed revisions to DCD Revision 17, the specification regarding foundation settlement adequately addressed the settlement requirement for the AP1000 nuclear island foundation and adjacent structures, and procedures for COL applicants to follow if predicted settlement exceeds the design limits. The staff finds this acceptable.

- The information provided by the applicant in the DCD on subsurface uniformity is reasonable, and the site investigation and site foundation material evaluation criteria are acceptable because they acknowledge that site parameter information is required to satisfy the design and regulation. The staff finds this acceptable

In summary, the staff finds that the changes to AP1000 DCD Tier 1, Table 5.0-1, and DCD Tier 2, Section 2.5.4, adequately describe the site-specific geotechnical and geophysical information and investigations that a COL applicant referencing the AP1000 DCD must provide to determine the properties and stability of all soils and rock that may affect the safety of nuclear power plant facilities, under both static and dynamic conditions, including the vibratory ground motions associated with the SSE. The staff concludes that the geological, seismological, and geotechnical engineering-related site parameters presented in Tier 1, Table 5.0-1, as well as in Tier 2, Table 2-1, are acceptable, because they meet the requirements of GDC 2, 10 CFR 52.47(a)(1), and 10 CFR 52.47(a)(2)(iv).

The applicant submitted changes to the DCD that provide the seismic design and supporting analysis for a range of soil conditions representative of expected applicants for a COL referencing the AP1000 design. These changes provide increased standardization for this aspect of the design. In addition, these changes reduce the need for COL applicants to seek departures from the current AP1000 design, since many sites do not conform to the currently-approved hard rock sites. Therefore, the change increases standardization and meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

2.5.5 Combined License Information for Stability and Uniformity of Slopes

The applicant made no additions or changes to DCD Section 2.5.5 from the certified design of Revision 15 of the DCD; therefore, the staff did not reevaluate any of the previously certified information in this section.

2.5.6 Combined License Information for Embankments and Dams

The applicant made no additions or changes to DCD Section 2.5.6 from the certified design of Revision 15 of the DCD; therefore, the staff did not reevaluate any of the previously certified information in this section.

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Westinghouse Electric Company, LLC (Westinghouse or the applicant) has submitted information in support of its design certification (DC) amendment application that it considers "proprietary" within the meaning of the definition provided in Title 10 of the Code of Federal Regulations (10 CFR) 2.390(b)(5), "Public inspections, exemptions, requests for withholding." The applicant has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information will appear as a blank space surrounded by "square brackets" as follows:

[ ]

The complete text of this chapter, including proprietary information, can be found at Agencywide Documents Access and Management System (ADAMS) Accession Numbers ML112091879. This document can be accessed by those who have specific authorization to access the applicant's proprietary information.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

Revisions 16 and 17 of the AP1000 Design Control Document (DCD) include a number of changes to Section 3.2.1, Tables 3.2-2 and 3.2-3 as well as related Chapter 17 changes for quality assurance (QA) requirements. The change to Section 3.2.1 is limited to a clarification regarding reference to 10 CFR 50.34, "Contents of applications; technical information," rather than 10 CFR Part 100, "Reactor site criteria." The change to Table 3.2-2 consists of the inclusion of notes to clarify the non-seismic (NS) classification of certain structures described in other DCD sections. The changes to the Table 3.2-3 primarily involve the addition of components and their seismic classifications.

3.2.1.1 Evaluation

The staff reviewed Revisions 16 and 17 of the DCD according to the guidance in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.2.1, "Seismic Classification," which references Regulatory Guide (RG) 1.29, , "Seismic Design Classification," Revision 4; RG 1.143, , "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2; RG 1.151, "Instrument Sensing Lines," Revision 1; and RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, for seismic classification of various structures, systems, and components (SSCs). As identified in 10 CFR 52.47, "Contents of applications; technical information," the application is based on regulatory guide revisions that were in effect 6 months before the docket date of the initial application. An NRC audit of design specifications performed October 13-17, 2008, for risk-significant components was also considered relative to seismic classification. The staff reviewed related technical reports (TRs) and also reviewed the nonsite-specific SSCs included in DCD Section 3.2.1 to determine if the scope was essentially complete.

The staff determined that the Section 3.2.1 change referencing 10 CFR 50.34 rather than 10 CFR Part 100 was acceptable, since 10 CFR 50.34 is referenced in the definition of the term safety-related in addition to 10 CFR Part 100. Both regulations provide similar acceptance criteria for offsite doses. The other DCD changes were primarily intended to resolve staff questions on the regulatory treatment of nonsafety systems (RTNSS). The staff determined that the clarifying notes to Table 3.2-2 were acceptable on the basis that structures designated as NS have augmented seismic requirements described in other DCD sections.

The staff's review of the DCD classification changes for RTNSS determined that, in general, the specific changes identified in the amendment are acceptable, but during the review of Revision 16, the staff identified several potential errors and omissions in a number of technical areas that needed clarification in the DCD. The staff reviewed Revision 17 to determine if the issues identified during the Revision 16 review could be closed. The staff's review evaluated the DCD changes to determine if it was appropriate to resolve these errors and omissions and these are discussed below under each topic. The technical review and resulting requests for additional information (RAIs) are not considered to represent new NRC requirements, but are intended to clarify statements in the DCD and address omissions in the application that have not been reviewed in the DC.

Augmented Seismic Requirements for RTNSS SSCs (RAI-SRP3.2.1-EMB2-01)

To comply with 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 2,, "Design Basis for Protection Against Natural Phenomena," SSCs important to safety are to be designed to withstand earthquakes. RTNSS SSCs that are important to safety but not specifically considered safety-related need not be classified as seismic Category I, but do require additional seismic design considerations under the RTNSS process to enable them to withstand earthquakes and meet GDC 1. The extent to which non-safety-related SSCs are seismically qualified is defined by the RTNSS process.

In DCD Revisions 16 and 17, a number of changes were made to the classification of SSCs including classification Table 3.2-3; and the changes in Revision 17 include previously omitted SSCs important to safety, such as the ancillary diesel generators and portions of the fire protection system (FPS).

The inclusion of the ancillary diesel generators reflects a Revision 16 RAI response defining additional seismic requirements for this RTNSS equipment to be located within buildings designed to Uniform Building Code (UBC) seismic requirements with additional requirements designated in some cases. DCD Section 8.3.1.1.3 identifies that the ancillary diesel generators and the fuel tanks are located in the portion of the Annex Building that is a seismic Category II structure. This location is acceptable because the supplemental seismic treatment does meet minimum requirements defined in the staff requirements memorandum (SRM) dated June 23, 1997, concerning SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," for equipment needed post-72 hour to be located such that there are no spatial interactions with any other nonseismic SSCs. On the basis of the SRM, no dynamic qualification of active equipment is necessary for SSCs needed for post-72 hour actions and staff considers equipment location in a seismic Category II building with seismic Category II anchorage to be acceptable. The RAI response also indicated that the seismic classification of SSCs is considered to be complete, but if design finalization identifies changes, the design change process should identify changes that would impact the detailed application of the classification to systems and components.

Although the standpipe portions of the FPS that are inside the reactor containment and auxiliary building are designated in DCD Table 3.2-3 as NS, comments in the table stipulate a seismic analysis consistent with American Society of Mechanical Engineers (ASME) Code Section III Class 3 systems. The staff finds this to be acceptable, since this meets the criteria for seismic analysis identified in NUREG-0800 Section 9.5.1 and RG 1.189 for portions of FPSs.

It was still not clear what additional seismic requirements may apply to certain Class D systems and components. DCD Section 3.2.2.6 states that, in regard to Class D, the systems and components are not designed for seismic loads. For example, other than anchorage, the seismic requirements for the ancillary diesel generators and other equipment to ensure their functionality following a seismic event is not defined. The staff’s guidance in a memorandum dated July 18, 1994, pertaining to AP600, identified a proposed review approach for equipment designated as important by the RTNSS process. Although a dynamic qualification test may not be necessary for this equipment, the SRM identified an approach where a dynamic analysis or qualification of electrical and mechanical equipment by experience may be used on a case-by-case basis. Staff is concerned that seismic anchorage alone does not ensure functionality of electrical and mechanical equipment following a safe-shutdown earthquake (SSE), unless it is supported by an analysis or experience. This concern was identified during the Revision 16 review as Open Item OI-SRP3.2.1-EMB2-01.

In an attempt to resolve this Revision 16 open item, the staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the onsite review are documented in the NRC report dated March 17, 2009. The applicant responded to Open Item OI-SRP3.2.1-EMB2-01 by referencing SECY-96-128 and NUREG-1793, “Final Safety Evaluation Report [FSER] Related to Certification of the AP1000 Standard Design,” Section 22.5.6, but DCD Section 3.2.1 was not updated to identify the basis cited in the response. The applicant believes that the guidance in the SRM dated July 18, 1994, is not applicable to the AP1000 DC review and the seismic design requirements imposed on components, identified as important by the RTNSS process, as identified in the AP1000 DCD in Table 3.2-3 and Westinghouse Commercial Atomic Power (WCAP)-15985, “AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process,” Revision 2, provide an appropriate level of seismic protection. The response further clarified that the design provides an alternate way of supporting long term operation of passive features using offsite supplied equipment that is independent of these RTNSS SSCs. Therefore, the applicant stated that there is no need to raise the level of seismic design requirements for these RTNSS SSCs to seismic Category I and concluded that the application of seismic Category II anchorages identified in DCD Table 3.2-3 will provide reasonable assurance that the SSCs identified by the RTNSS process as important for the post 72-hour operation are functional in the required time frame, even after the most limiting design basis earthquake.

The staff reviewed the basis for seismic requirements applicable to certain RTNSS SSCs cited in the response. SECY-96-128 and the associated memorandum referenced in the response is applicable to AP600 and states that the site be capable of sustaining all design basis events with onsite equipment and supplies for the long term. The equipment required after 72 hours need not be in automatic standby response mode, but must be readily available for connection and be protected from natural phenomena including seismic events (pursuant to GDC 2).

Therefore, staff disagrees with the applicant's position that offsite equipment may be credited for equipment needed post-72 hours. However, based on staff guidance, no dynamic qualification of this equipment is necessary and equipment is to be designed with seismic Category II anchorage and located within a seismic Category II structure.

Although the approach proposed in the SRM dated July 18, 1994, is applicable to AP600 rather than AP1000, this document proposed a review approach for RTNSS systems in passive designs where nonsafety-related systems designated to be important by the RTNSS process (IRP) are needed to perform their required function after an earthquake. For example, IRP systems and components should not be required to be classified as seismic Category I, but staff may consider the use of experience data for seismic qualification on a case-by-case basis. The SRM dated June 23, 1997, regarding SECY-96-128 for AP600, clarified a staff position that post-72 hour SSCs need not be safety-related, but equipment anchorages must be consistent with the SSE design equipment anchorages of seismic Category I items and there should be no adverse interactions. Further, this memorandum clarified that no dynamic qualification of active equipment is necessary. Although operability or functionality is not entirely ensured unless either classified as seismic Category I or otherwise justified, it is reasonable to expect that seismic Category II anchorage and location within a seismic Category II structure will afford some degree of structural integrity. Therefore, staff accepts the applicant's position that the seismic classification is basically consistent with previous positions for AP600 documented in documents related to SECY-96-128 and NUREG-1793. As a result of this review, Open Item OI-SRP3.2.1-EMB2-01 is closed.

Scope (RAI-SRP3.2.1-EMB2-02)

During the review of Revision 16, the staff was concerned that the scope of SSCs identified in DCD Section 3.2.1 does not appear to be complete and this was identified as an open item. In RAI-SRP3.2.1-EMB2-02, the applicant was requested to identify the seismic classification of any nonsite-specific SSCs, such as the circulating water system (CWS), electrical items, and reactor vessel insulation, within the scope of the DCD that are not included in the DCD tables.

The RAI response clarified that Table 3.2-3 does not include information on electrical, instrumentation or architectural elements and identified that Table 3.2-2 will be revised to include seismic requirements for various structures and that Table 3.2-3 will be revised for the FPSs. The response also clarified that, although the design of some of the SSCs is the responsibility of the combined license (COL) applicant, the seismic categorization is provided as part of the DC. The response identified the CWS and raw water system (RWS) as NS.

The staff reviewed Revision 17 and determined that the changes do not entirely resolve the staff's concerns. Relative to completeness of scope in the application, the applicant included the omitted ancillary diesel generators and the FPS components in the DCD and references DCD Section 3.7.2.8 for seismic requirements applicable to NS structures. However, the seismic classification of the CWS and RWS identified in the RAI response is not included in the revised DCD tables. Similarly, DCD Revision 17 does not include the seismic classification for the electrical and instrumentation components or other miscellaneous SSCs such as the reactor pressure vessel (RPV) insulation. This concern was identified during the review of Revision 16 as Open Item OI-SRP3.2.1-EMB2-02.

In an attempt to resolve this Revision 16 open item, staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the onsite review are documented in the NRC report dated March 17, 2009. The applicant responded to Open Item OI-SRP3.2.1-EMB2-01 by revising the DCD, referencing DCD Table 3.11-1 for seismic classification of electric and instrumentation equipment and stating that the detail for seismic classification in the AP1000 DCD is sufficient for DC. The revised DCD includes RPV insulation as seismic Category II and additional components, such as valves, the secondary core support structure and components associated with the reactor coolant system (RCS).

The staff reviewed the applicant's response. The response adequately justifies that the seismic classification of electrical items need not be included in Table 3.2-3 since they are outside the scope of NUREG-0800 Section 3.2.1, and the classification of these items in Table 3.11-1 as seismic Category I should be sufficient to support the seismic review of electric items addressed in Chapter 8. Although the response does not revise DCD Table 3.2-3 and piping and instrumentation drawings (P&IDs) to include the seismic classification of all SSCs, such as piping, other sections of the DCD do identify seismic classification for piping systems and specific equipment. It is understood that the interconnected piping has a seismic classification similar to that of equipment and components. The seismic classification of SSCs added in Table 3.2-3 is consistent with RG 1.29 and GDC 2. Therefore, the staff concludes that, although the scope of SSCs seismically classified in Table 3.2-3 is not complete, other sections of the DCD include the seismic classification of SSCs not included in Table 3.2-3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. As a result of this review, Open Item OI-SRP3.2.1-EMB2-02 is closed.

Augmented Quality Assurance (QA) Requirements for Seismic Category II SSCs (RAI-SRP3.2.1-EMB2-03)

In Revision 16 DCD Section 3.2.1.1.2 was revised to reference DCD Section 17.5 rather than Section 17.4 for the COL QA requirements for seismic Category II SSCs. During the review of Revision 16, the staff determined that DCD Table 3.2-3 included in Revision 16 did not identify specific augmented QA requirements that apply to seismic Category II SSCs. The staff was concerned that Section 3.2, Table 3.2-3 or Chapter 17 included in DCD Revision 16 do not adequately define specific augmented QA requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," for seismic Category II SSCs. It was not clear if the COL applicant is to provide these requirements for the procurement of nonsite-specific SSCs. In RAI-SRP3.2.1-EMB2-03, the applicant was requested to clarify to what extent the pertinent QA requirements of Appendix B to 10 CFR Part 50 apply to nonsite-specific seismic Category II SSCs and to identify the DCD section or other document that describes those requirements. The RAI response restated the DCD Section 3.2.1.1.2 statement that pertinent portions of 10 CFR Part 50, Appendix B apply to seismic Category II SSCs and that pertinent portions are those required to provide that unacceptable structural failure or interaction with seismic Category I items does not occur. The response further clarified that seismic Category II SSCs are covered by the same quality programs and procedures as seismic Category I and the extent of design activities are determined by the responsible engineers and are identified in the design specifications and design criteria documents.

The staff reviewed the changes included in Revision 17 and determined that neither DCD Section 3.2, Table 3.2-3 nor Section 17.5 has been revised to identify specific augmented QA requirements for seismic Category II SSCs. This concern was identified during the review of Revision 16 as Open Item OI-SRP3.2.1-EMB2-03.

In an attempt to resolve the Revision 16 open item, staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the onsite review are documented in the NRC report dated March 17, 2009. The applicant responded to Open Item OI-SRP3.2.1-EMB2-03 by stating that it does not agree that specific QA requirements for seismic Category II SSCs should be included in the DCD, but the DCD is to be revised to clarify that QA requirements are performed consistent with the applicant's quality plan as described in Section 17.3. In the response, the applicant identified three different approaches applied to nonsafety-related SSCs that are subject to seismic requirements and stated that AP1000 seismic Category II SSCs are subject to the AP1000 quality plan as described in NUREG-0800 Section 17.3 QA requirements.

In a subsequent response to the staff concerns, the applicant clarified its process to identify supplemental requirements for RTNSS SSCs and seismic Category II SSCs. The applicant stated that application of augmented QA is a function of the RTNSS assessment, not the seismic categorization. The response identifies that the Design Reliability Assurance Program (D-RAP) described in DCD Section 17.4 does not impose augmented design or quality requirements on SSCs and that DCD Table 3.2-1 includes adequate reference to seismic Category II design and quality requirements. The response recognizes that DCD Section 3.2.2.6 does not specifically allow for the use of pertinent portions of 10 CFR Part 50 Appendix B to seismic Category II applications and proposes a DCD revision for clarity.

Although the applicant does not impose quality requirements based on the D-RAP, the staff believes that reliability depends on the design and quality of the SSCs and that the purpose of the D-RAP is to ensure reliability using the design process. As stated in DCD Section 17.4, the AP1000 D-RAP is implemented as an integral part of the AP1000 design process to provide confidence that reliability is designed into the plant. NUREG-0800 Section 17.4 also states that the objective of the reliability assurance program (RAP) is to ensure that the reliability is properly considered and designed into the plant. Draft DC/COL-ISG-018, "Interim Staff Guidance on NUREG-0800 Standard Review Plan Section 17.4, 'Reliability Assurance Program,'" further states that the purpose of the RAP is that the reactor is designed consistent with key assumptions (including reliability) and key insights. During the DC phase, the applicant prepared details of the D-RAP and implemented appropriate graded controls related to design activities for nonsafety-related within the scope SSCs. Those supplemental requirements/graded controls (special treatment) for risk-significant SSCs may include short term availability controls, design requirements, seismic requirements, inspections, maintenance, or QA controls to ensure reliability. One of the design considerations in the AP1000 D-RAP is that the design reflects the reliability values assumed in the design and probabilistic risk assessment (PRA) as part of procurement specifications. DCD Sections 3.2.1.1.2 and 3.2.2.6 are to be revised to reference DCD Section 17.3 for augmented quality requirements for seismic Category II SSCs consistent with RG 1.29, without a specific reference to the D-RAP. The staff recognizes that the RTNSS process combined with the D-RAP should be used to establish reliability of risk-significant SSCs so that appropriate specific QA requirements may be established during the detailed design. Therefore, it is reasonable to expect appropriate QA requirements to be applied to risk-significant seismic Category II SSCs and that these requirements are to be included in the design or procurement specifications that can be verified when available. As a result, Open Item OI-SRP3.2.1-EMB2-03 is closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

List of SSCs Needed for Continued Plant Operation

10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” Section IV(a)(2)(I), states that SSCs necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits when subject to the effects of the operating basis earthquake (OBE) ground motion. NUREG-0800 Section 3.2.1 states that, if the applicant has set the OBE ground motion to the value one-third of the SSE ground motion, then the applicant should also provide a list of SSCs necessary for continued operation that must remain functional without undue risk to the health and safety of the public and within applicable stress, strain and deformation limits, during and following the OBE. AP1000 DCD Section 3.7 states that the OBE for shutdown is considered to be one-third of the SSE.

10 CFR Part 50, Appendix S, Section IV(a)(3), states that if vibratory ground motion exceeding that of the OBE ground motion or if significant plant damage occurs, the licensee must shut down the nuclear power plant, and that, prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained. Identification of the necessary SSCs and inclusion of the equipment at the appropriate seismic classification level in the DCD would allow the plant to address the requirements when the need exists.

In an attempt to obtain this information, staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a safety conclusion. The results of the onsite review are documented in the NRC report dated March 17, 2009.

In RAI-SRP3.2.1-EMB2-06, the applicant was requested to provide this list of SSCs necessary for continued operation or an alternative to address the requirements. The applicant was requested to include in the DCD the list of SSCs necessary for continued operation. This concern was identified as Open Item OI-SRP3.2.1-EMB2-06.

The applicant’s response to Open Item OI-SRP3.2.1-EMB2-06 clarifies that the SSCs necessary to protect the public health and safety are the safety-related SSCs identified in Section 3.2.2 of the DCD and tabulated in DCD Table 3.2-3. The response does not address nonsafety-related SSCs that may be important to safety, such as RTNSS SSCs, but the applicant identifies that the capability of nonsafety-related SSCs to support power production following an OBE is an investment protection issue. The response further identifies that post earthquake planning is the responsibility of the operators and is not included in the design certification. The applicant proposes a revision to DCD Section 3.2.1.1 to add a statement regarding the safety-related SSCs in regard to 10 CFR 50 Appendix S. In response to further staff concerns relative to pre-earthquake planning and RG 1.166, “Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions,” applicability, the applicant revised its response to clarify that pre-earthquake planning is the responsibility of the COL holder and that DCD Section 3.7.5.2 identifies a COL information item for post-earthquake procedures. The response stated that post-earthquake procedures will follow Electric Power Research Institute (EPRI) guidance and it was noted that the COL applicant would be able to address RG 1.166 and the list of SSCs to be included in procedures.

The staff agrees that RG 1.166 is not applicable to the DC and post-earthquake planning is the responsibility of the operators and not included in the DC. Therefore, this is considered to be addressed in the procedures developed by the COL applicant. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.2.1.2 Conclusion

The seismic classification of SSCs is, in general, consistent with RG 1.29, with the exceptions identified in DCD Appendix 1A.

Therefore, on the basis of its review of DCD Revision 19 included in Tier 2 Section 3.2.1, Tables 3.2-2 and 3.2-3, the staff concludes that the AP1000 safety-related SSCs, including their supports, are properly classified as seismic Category I, in accordance with Position C.1 of RG 1.29. In addition, the staff finds that DCD Tier 2 includes acceptable commitments to Positions C.2, C.3, and C.4 of RG 1.29. This constitutes an acceptable basis for satisfying, in part, the portion of GDC 2 that requires that all SSCs important to safety be designed to withstand the effects of natural phenomena, including earthquakes.

3.2.2 Quality Group Classification

Revisions 16 and 17 of the DCD include a number of changes to Section 3.2.2 and Table 3.2-3 related to the AP1000 classification system and to Chapter 17 for QA requirements. The changes to Section 3.2.2 include a clarification regarding reference to 10 CFR 50.34 rather than 10 CFR Part 100 and clarifications regarding applicability of ASME Code Section III to pressure-retaining components. The changes to the Table 3.2-3 primarily involve the addition of components and their AP1000 classifications.

3.2.2.1 Evaluation

The staff reviewed the DCD Revisions 16 and 17 according to the guidance in NUREG-0800 Section 3.2.2, “Quality Group Classification,” which references RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” for quality group (QG) classification of various SSCs. The staff review considered that additional detailed design information needed to be verified. An NRC audit of design specifications performed October 13-17, 2008, for risk-significant components was also considered relative to QG classification. The staff also reviewed TR-103, “Fluid System Changes,” APP-GW-GLN-019, Revision 2, and TR-106, “AP1000 Licensing Design Changes for Mechanical System and Component Design Updates,” APP-GW-GLN-106, Revision 1, which address various system changes that could have an impact on QG classifications.

The staff determined that the DCD Section 3.2.2.1 change referencing 10 CFR 50.34 rather than 10 CFR Part 100 was acceptable since 10 CFR 50.34 as well as 10 CFR Part 100 are referenced in the definition of the term safety-related. Both regulations provide similar acceptance criteria for offsite doses. The other DCD changes were primarily intended to resolve staff questions on RTNSS. The staff also determined that the clarifying notes concerning applicability of ASME Code Section III to pressure boundary components were acceptable with the understanding that ASME Code Section III also applies to supports for pressure boundary systems and components.

The staff’s review of the DCD changes determined that, in general, the specific changes identified in the application are acceptable, but that several potential errors and omissions in a number of technical areas need clarification in the DCD. During the Revision 16 review, the staff prepared RAIs to resolve these errors and omissions and these are discussed below under each topic. The technical review and resulting RAIs are intended to clarify statements in the DCD and address omissions in the application.

Supplemental Requirements for Nonsafety-Related Passive SSCs Important to Safety (RAI-SRP3.2.2-EMB2-01)

During the review of Revision 16, the staff was concerned that neither DCD Section 3.2 nor Table 3.2-3 adequately defines specific supplemental quality standards and QA programs applied to nonsafety-related passive SSCs that are important to safety and risk-significant. In RAI-SRP3.2.2-EMB2-01, the applicant was requested to clarify what supplemental quality standards and QA program are applied to nonsafety-related passive SSCs that are important to safety.

The RAI response clarified that codes and standards for Class D systems and components provide an appropriate level of integrity and functionality. The response also stated that the PRA did not identify SSCs that need a more rigorous code or standard than those identified in the DCD to provide improved reliability.

The staff reviewed the applicant's response to RAI-SRP3.2.2-EMB2-01 and determined that the response partially resolves its concerns. Although the PRA and RTNSS process did not apparently identify any supplemental requirements for passive components, the staff is concerned that supplementation may be appropriate, especially where there is insufficient operating history. For example, where high density polyethylene (HDPE) piping is to be used for underground plant service water system (SWS) piping that is considered a risk-significant defense in depth RTNSS system, additional special treatment should be imposed on design and QA requirements to ensure its integrity consistent with the system's safety function. Special treatment is appropriate for buried non-metallic piping that does not have a sufficient operating history in similar applications where failures are possible, unless special precautions are taken during design, fabrication, installation, and testing. Examples of supplementation applied to important to safety HDPE piping are addressed in ASME Code cases and relief requests. Although the plant service water piping is not considered safety-related, it is important to safety and GDC 1, "Quality Standards and Records," requires that, where generally recognized codes and standards are used, they shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Therefore, passive SSCs used in risk-significant RTNSS systems, such as the SWS piping should be supplemented or modified accordingly. This concern was identified during the Revision 16 review as Open Item OI-SRP3.2.2-EMB2-01.

In an attempt to resolve the Revision 16 open item, the staff performed an onsite review to examine detailed design documents that could define the additional information for staff to reach a reasonable safety conclusion. The results of the onsite review are documented in the NRC report dated March 17, 2009.

The applicant's response to Open Item OI-SRP3.2.2-EMB2-01 clarified that, in regard to HDPE piping used in the SWS, which is identified as a RTNSS system, HDPE will only be used in flow paths that are not required to support the important-to-safety function of decay heat removal. Therefore, the applicant concluded that supplementation or modification to meet GDC 1 is not required in the application of HDPE piping in the SWS.

The staff agrees that, if HDPE is only used in portions of the SWS that are not risk-significant, supplementation or modification to ensure reliability of HDPE need not be identified. However, the staff was concerned that supplementation or modification of other risk-significant passive SSCs has not been identified. The applicant's revised response clarified that the RTNSS process is independent of the D-RAP and the D-RAP does not impose supplementation as a requirement. However, the response identifies that RTNSS SSCs apply augmented QA in accordance with DCD Table 17-1, "QA Requirements for SSCs Important to Investment Protection." These QA requirements and scope of SSCs included in the D-RAP for RTNSS SSCs are reviewed in other sections of this safety evaluation report (SER) according to NUREG-0800 Section 17.4 and draft DC/COL-ISG-018. Although the applicant does not impose quality requirements based on the D-RAP, the staff's opinion is that reliability depends on the design and quality of the SSCs and that the purpose of the D-RAP is to ensure reliability using the design process. As stated in DCD Section 17.4, the AP1000 D-RAP is implemented as an integral part of the AP1000 design process to provide confidence that reliability is designed into the plant. NUREG-0800 Section 17.4 also states that the objective of the RAP is to ensure that reliability is properly considered and designed into the plant. DC/COL-ISG-018 concerning the D-RAP and implementing appropriate graded QA controls further states that the purpose of the RAP is to assure that the reactor is designed consistent with key assumptions (including reliability) and key insights. Supplemental requirements/graded controls (special treatment) for risk-significant SSCs may include short term availability controls, design requirements, seismic requirements, inspections, maintenance, or QA controls to ensure reliability.

One of the design considerations in the AP1000 D-RAP is that the design reflects the reliability values assumed in the design and PRA as part of procurement specifications. To be consistent with the Interim Staff Guidance (ISG), the application should specify the QA controls related to DC design activities in accordance with the provisions in Part V, "Non-safety-related SSC Quality Controls," of NUREG-0800 Section 17.5 for the nonsafety-related, within the scope of SSCs. Based on the ISG, the NRC verifies the DC applicant's D-RAP, including its implementation during the DC application phase, through the agency's safety evaluation review process, as well as audits. Therefore, the staff recognizes that the supplementation needed to ensure reliability assumed in the PRA is to be determined by the RTNSS process combined with the D-RAP and that the inspection, test, analyses, and acceptance criteria (ITAAC) in Table 3.7-3 of Tier 1 of the AP1000 DCD have been developed to allow review of this process. As a result, Open Item OI-SRP3.2.2-EMB2-01 is closed.

Application of Unendorsed ANS Standard (RAI-SRP3.2.2-EMB2-02)

DCD Revision 16 added American Nuclear Society (ANS) Standard 58.14-1993, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors," as a reference for safety classifications and this standard continues to be referenced in Revision 17. The staff was concerned that withdrawn and outdated ANS 58.14-1993 is not NRC-endorsed and cannot be used as a basis for acceptability of classifications. In RAI-SRP3.2.2-EMB2-02, the applicant was requested to either reference an updated classification standard or adequately describe the classification criteria in the application.

In its response, the applicant clarified that the referenced documents provide background for the equipment classification, but the AP1000 classification approach does not rely on the endorsement of any particular standard as the basis of the classification approach.

The staff reviewed the applicant's response and concludes that, although the referenced classification standard is being included in the DCD, the staff will not rely on this standard or other unendorsed standards as a basis for acceptability of classifications. On this basis, RAI-SRP3.2.2-EMB2-02 is closed.

Codes and Standards (RAI-SRP3.2.2-EMB2-03)

The SRM dated July 31, 1993, concerning SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," identified that the staff will review passive plant designs using the newest codes and standards endorsed by the NRC and unapproved revisions to the codes and standards referenced in the DCD will be reviewed on a case-by-case basis. During the Revision 16 review, the staff was concerned that editions of codes and standards referenced in the DCD not otherwise endorsed by the NRC might not be current. In RAI-SRP3.2.2-EMB2-03, the applicant was requested to clarify which editions of various codes and standards were NRC-endorsed and to clarify if current editions of codes and standards would be applied to the detailed design and procurement of AP1000 SSCs so that these codes and editions might be reviewed on a case-by-case basis.

The applicant clarified that codes and standards were generally those in effect six months prior to the submittal of the application and these editions would be applied to the detailed design and procurement of AP1000 SSCs. The response identified that, in a limited number of cases, the applicant was updating the revisions of codes and standards and this change would be specifically identified in a DCD revision.

The staff agreed that editions of codes and standards in effect six months prior to the application are acceptable and noted that the staff will have the opportunity to review future changes. DCD Section 3.2.6, Revision 17 made no changes to the referenced codes and standards editions and on this basis RAI-SRP3.2.2-EMB2-03 is closed.

Classification of Fire Protection System (RAI-SRP3.2.2-EMB2-04)

During the DCD Revision 16 review the staff was concerned that DCD Section 3.2.2.7 had been revised to identify that both Class F and G are used for FPSs, but Table 3.2-3 did not identify FPS SSCs that were classified as Class F and G. The staff was concerned that the classification of the FPS in DCD Revision 16 was not complete and in RAI-SRP3.2.2-EMB2-04, the applicant was requested to submit the classifications for the entire FPS.

In the RAI response, the applicant submitted a revised Table 3.2-3 for additional FPS piping and components. The staff concurs that inclusion of the revised DCD Table 3.2-3 represents a generally complete scope of FPS piping and components and that the classification of these as consistent with RG 1.29 and NUREG-0800 Section 9.5.1 criteria is an acceptable regulatory basis. The classification of the standpipe system as AP1000 Class F constructed to American National Standards Institute (ANSI) B31.1, "Code for Pressure Piping," and categorized as NS with a seismic analysis consistent with ASME Code Section III Class 3 is consistent with the guidance in NUREG-0800 Section 9.5.1 and RG 1.189 (considered not applicable to AP1000) and is, therefore, acceptable. Therefore, RAI-SRP3.2.2-EMB2-04 is closed.

3.2.2.2 Conclusion

On the basis of its review of the DCD Amendment Section 3.2.2, and the above discussion, the staff concludes that the QG classifications of the important to safety pressure-retaining fluid systems and their supports, as identified in DCD Tier 2, Tables 3.2-1 and 3.2-3, and related P&IDs in the DCD, are consistent with RG 1.26, other than exceptions identified in DCD Appendix 1A, and are acceptable. These tables and P&IDs identify major components in fluid systems (i.e., pressure vessels, heat exchangers, storage tanks, piping, pumps, valves, and applicable supports). In addition, P&IDs in the DCD identify the classification boundaries of interconnecting piping and valves. All of the above SSCs will be constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26 as described above and applicable ASME Codes and industry standards provide assurance that component quality will be commensurate with the importance of the safety functions of these systems. Therefore, the staff concludes that the application meets the requirements of GDC 1 for QG classifications.

3.3 Wind and Tornado Loadings

3.3.1 Summary of Technical Information

With regard to wind and tornado loads on the seismic Category I structures, the AP1000 DCD, Revision 17 changes the shield building by reducing its height by 1.52 meters (m) (5 feet (ft)). As a result, the wind and tornado loads are also altered.

3.3.2 Combined License Information 3.3-1 and 3.5-1

The commitment to address combined operating and licensing information (DCD COL Information Items 3.3-1, “Wind and Tornado Site Interface Criteria,” and 3.5-1, “External Missile Protection Requirements,” concerning site interface criteria for wind and tornado by the COL applicant) is defined in TR-5, “AP1000 Wind and Tornado Site Interface Criteria,” APP-GW-GLR-020, Revision 4. Revision 17 of the DCD includes the following applicable changes:

- Evaluation of generic wind and tornado loadings on structures;
- Provision of the plant specific site plan and comparison with the typical site plan shown in Figure 1.2-2, “Site Plan,” of DCD Section 1.2;
- Discussion of missiles produced by tornadoes and other external events; and
- Evaluation of other buildings for collapse and missile generation.

Based on the above mentioned evaluations, the applicant is to demonstrate that any exceedances or differences in the evaluation results from what is specified in the DCD will not compromise the safety of the nuclear power plant.

3.3.3 Evaluation

The shield building is a seismic Category I structure located on the nuclear island (NI). The development of loads on the air baffle in the top portion of the shield building due to the design-basis wind and tornado is a safety concern. The methodology for load evaluation follows the AP600 approach combined with wind tunnel testing, which gives rise to the wind loads across the air baffle, assuming a constant tornado wind speed with the height of the building. This means that the total wind load on the structure increases with increasing height of the building. The proposed change to the DCD includes a 1.52 m (5 ft) reduction of the total height of the shield building. As a result, total wind loads applied to the building are altered. This alteration may influence important design parameters.

The staff reviewed the change with regard to the impact on the wind load to determine its acceptability. Since the wind loads are in direct proportion to the height of the structure, the total net load applied to the building will be less than before the change. This means that, for a fixed diameter, a reduction of 1.52 m (5 ft) in height will result in approximately 2.5 percent reduction in the wind loads applied to the building. The outcome of this change of design is an increase in safety margin due to decreasing applied loads. Thus, the design change increases the degree of conservatism and is, therefore, acceptable. The staff concludes that the application meets the requirements of GDC 2.

3.3.4 Development of COL Information Items

The DCD Revision 17 via TR-5, Revision 4 provides the detailed requirements specified in COL Information Items 3.3-1 and 3.5-1. In order to close out the COL Information Items 3.3-1 and 3.5-1, the following items must be addressed by the COL applicant:

With regard to site interface criteria for wind and tornado (Information Item 3.3-1), the DCD states:

> The site parameters wind speeds for which the AP1000 plant is designed are given in Table 2-1, “Site Parameters (Sheets 1 - 4), of the DCD. In addition, the design parameters applicable to tornado are given in DCD Section 3.3.2.1, including maximum rotational speed of 240 mph (385 km/h); max. translational speed of 60 mph (96 km/h); radius of max. rotational wind from center of tornado, 150 ft (45-3/4 m); atmospheric pressure drop of 2.0 psi (13.8 kPa) and rate of pressure change of 1.2 psi per sec (8.3 kPa per sec). Should the site parameters exceed those bounding conditions; the applicant will be required to demonstrate that the design conforms to the acceptance criteria.

DCD Section 3.3.3, “Combined License Information,” includes only the commitment that COL applicants referencing the AP1000 certified design will address site interface criteria for wind and tornado loadings. This change via TR-5 provides specific interface criteria, including necessary information items for the COL applicant. The COL information items include: development of site-specific parameters, verifications of bounding conditions, plant layout and site arrangement. Should the site parameters exceed those bounding conditions, the applicant will be required, either through analysis, testing or combined analysis and testing, to demonstrate that the design conforms to the acceptance criteria.

The staff reviewed the interface criteria for wind and tornado provided in TR-5 including evaluation of generic wind and tornado loadings on structures; discussion of missiles generated by tornadoes and extreme winds, and evaluation of missile generation and effects of building collapse on NI structures. Examination of those criteria revealed that they are necessary and sufficient in providing appropriate input to the design of safety-related SSCs. These COL

Information Items are deemed to show compliance with the Commission's regulations including GDC 2 in Appendix A to 10 CFR Part 50, and thus are acceptable.

With regard to tornado-initiated building collapse (Information Item 3.3-1) the DCD states:

> If the COL applicant has adjacent structures different from the typical site plan shown in Figure 1.2-2 of the DCD Section 1.2, a justification must be provided to show that they will not collapse, or their failure will not impair the structural integrity of the nuclear island safety-related structures. Now, the structures in the typical site plan have been evaluated for tornado-initiated failure or collapse. The analysis showed that they will not compromise the safety of the nuclear island structures or their seismic categories reclassified.

The staff reviewed the analysis and found it technically sound, except for one issue that requires further investigation. The radwaste building was evaluated for its potential collapse on the NI, demonstrating that it would not impair the structural integrity of the NI safety-related structures (see DCD Section 3.7.2.8.2, "Radwaste Building"). However, because of the addition of 3 liquid radwaste monitor tanks (see TR-106), which completely alters the structural dynamic characteristics of the building; it is not clear whether this conclusion is still valid. The staff reviewed the applicant's response to RAI-SRP3.7.2-SEB1-02, Revision 1, dated October 1, 2008, and determined that it was not acceptable because the staff's calculation of the maximum kinetic energy calculated using Method 3 in DCD Section 3.7.2.8.2 (6.8×10^7 joules (J) or 6.0×10^8 inch-pounds (in-lb)) for the water tank missile far exceeded that of the water tank missile (3.4×10^4 joules or 3.0×10^5 in-lb) claimed in the response. The staff's calculation was based on the assumptions adopted by the RAI response that the mass of a single water tank is 65,673 kilograms (kg) (144,781 pounds (lb))) and the velocity is 45.7 meters per second (m/s) (150 feet per second (fps) or 105 mph). This concern was identified as Open Item OI-SRP3.7.2-SEB1-02.

The applicant's approach to resolve the concern was to show that during a design-basis tornado event, the three water tanks will remain stationary, not result in a moving missile, then there would be no safety concern on the missile impact-induced damage to NI structures, and this open item could be closed. On May 13, 2010, the staff carried out an onsite audit on this report at the applicant's Twinbrook office. The safety analysis in APP-1000-CCC-007, "Further Evaluation of Potential Tornado Missiles on Nuclear Island," Revision 0, shows that during a tornado event with a design-basis wind speed of 134 m/s (300 mph), a total force of 12246 kilograms (kg) (27 kilopounds (kip)) will be produced by the tornado, and applied at each water tank, according to the American Society of Civil Engineers (ASCE) 7-98, "Minimum Design Loads for Buildings and Other Structures" that is acceptable to the NRC. Meanwhile, the six anchorage support bolts at each tank base were designed to resist a seismic force of up to 13607 kg (30 kip) based on the UBC. The conclusion was that because the applied tornado force on the tank is less than the resistance capability of the tank supports at the base, the tanks will remain stationary, and not become a damaging missile. The staff reviewed the calculations, and performed an independent confirmatory analysis using a new edition of ASCE 7-05, "Minimum Design Loads for Buildings and Other Structures," formula. The results showed that a tornado wind speed exceeding 141 m/s (316 mph) will break the anchor supports, resulting in high energy water tank missiles. Any wind speed higher than this limit will turn the tank into a missile, and therefore will not be acceptable. But because the design-basis tornado wind speed is only 134 m/s (300 mph) less than the limit with a safety margin of 5 percent, the water tanks will not become a moving missile. Based on the confirmatory analysis, the staff finds that the calculations provided by the applicant are acceptable. Thus, Open Item OI-SRP3.7.2-SEB1-02 is closed.

With regard to missiles generated by external events (COL Information Item 3.5-1) the DCD states:

> The AP1000 tornado missiles used for design are defined in Table 2.2-1 of the DCD Subsection 3.5.1.4 in terms of missile type vs energy spectrum, which is consistent with RG 1.76 (Reference 3). Other than tornado, missiles may be generated from external events such as transportation accidents or explosions. The COL applicant is responsible for identifying sources in the plant and the external events that could cause a producing missile to threaten the integrity of AP1000 safety-related SSCs. The missile energy should be compared with the Table in 3.5.1.4. If the external event missile has higher kinetic energy, the effect of the impact must be evaluated to show that it does not compromise the safety of the AP1000 safety-related structures.

In a letter dated December 23, 2008, the applicant responded to RAI-SRP3.3.2-SEB1-01 regarding the issue of missiles that are produced by the potential blow-off of the siding on the annex building as well as the turbine building. In its response, the applicant indicated that “The automobile in the missile spectrum included in the AP1000 would appear to bound the mass and energy of sheet metal siding. Also there are no safety-related structures, systems, and components outside of the Auxiliary Building and Shield Building. The walls of these buildings are reinforced concrete at least two feet thick. Tornado driven siding would not be expected to be a challenge to reinforced concrete walls.” The staff notes that the construction of the shield building is not reinforced concrete (RC) and can best be described as “steel-concrete-steel modular wall construction.” It is likely that the siding missile can penetrate the steel sheet of the modular wall of the shield building. The reanalysis of the shield building for a tornado-driven siding missile strike was identified as Open Item OI-SRP3.3.2-SEB1-01. An onsite audit meeting was held on February 24, 2010, at the applicant’s Twinbrook office where the penetration issue was discussed in detail based on the principles of mechanics in the areas of indentation, penetration and fracture. In a letter dated March 24, 2010, the applicant responded to Open Item OI-SRP3.3.2-SEB1-01 regarding the damage induced by siding missiles. In the response, it concluded that the penetration will be zero according to the basic assumptions, methodology and detailed calculations presented in APP-1000-CCC-007, Revision 0.

An onsite review of the report was performed by the staff on May 13, 2010 at the applicant’s Twinbrook office. The review reveals that there is a basic assumption in the analysis that all kinetic energy is converted to strain energy in the siding and the target wall or roof. The possibility of conversion to thermal energy or fracture energy is ruled out with no justification, and the penetration issue was not addressed. However, it is well-known that when two materials are brought into contact the harder material is bound to scratch or penetrate the softer material even if the velocity is very slow or buckling occurs at the high speed. Thus, as long as the hardness of the siding material is slightly higher than that of the building wall or the roof, a finite amount of penetration must occur. Indeed, in the confirmatory analysis performed by the staff, it was estimated, based on the data provided by the applicant on the siding missile, a penetration of about 2.54 centimeter (cm) (1 inch (in)) and 51 cm (20 in) will result from the impact on the steel panel and concrete roof respectively when steel siding weighing 7.8 kg (17.2 lb) travelling at a speed of 134 m/s (300 mph) makes a corner impact on the flat object. Those penetration depths were estimated using the appropriate formula given in NUREG-0800

Section 5.3.2 “Barrier Design Procedures.” There are no data available to confirm those estimates.

However, test data provided in a similar, but less severe, blast test carried out by J.R. McDonald using a timber plank missile travelling at 67 m/s (150 mph), weighing 6.8 kg (15 lb) with a 0.6 m by1.2 m (2 ft by 4 ft) contact area showed a penetration of 8.0 mm (5/16 in) for a steel panel and 15.2 cm (6 in) for a concrete slab. (References: (1) J.R. McDonald, “Impact Resistance of Common Building Materials to Tornado Missiles,” Journal of Wind Engineering and Industrial Aerodynamics,” Vol. 36, pp717-724, 1990; (2) M.K. Singhal and J.C. Walls: “Evaluation of Wind/Tornado-Generated Missile Impact,” in Table 3, ORNL Conference No. 9310102-18). Those data suggest that the penetration estimates using the NUREG-0800 Section 3.5.3 proposed formula are reasonable.

Given the potential local damage, a study was made in the confirmatory analysis to investigate whether the structural integrity of the NI structures would be compromised. First, from the geometry of the steel siding, those penetrations will produce a thru crack of 7.6-10.2 cm (3-4 in) long in the steel wall and up to 51 cm (20 in) long in the RC roof. It is important to note that the NI structure is under severe loads during a tornado event. The major loadings include a tornado wind load plus huge concentrated loads applied at a building location anywhere from grade to Elevation (El.) 293, resulting from the impacts by automobile missile strikes coming from the nearby raised parking lots (see Section 3.5.1.4). Thus, due to the resulting large bending moment created by the tornado loadings, tensile stress field is established in the structural components containing those flaws as the siding missile’s striking site is always located on the tensile side. In the worst-case scenario when the crack happens to be located in the critical section where the tensile stress is the maximum, it is possible, according to the principle of fracture mechanics, that the crack will immediately propagate unstably if the applied stress intensity factor (which is a function of the crack size, geometry and the applied stress), exceeds the toughness resistance of the material ~345 megapascal (MPa) (~50 kilopounds-force per square inch (ksi). Eventually the crack will be arrested in the compressive stress zone. Thus, potentially a crack several feet long with noticeable opening can result as a consequence of the local impact damage from the tornado missile strikes. However, because of the large dimensions of the structures, a total collapse of the building is not likely, due to the residual strength of the components (e.g., inner steel panel of the S-C wall or intact rebar in the RC roof). The structural integrity can still be maintained.

Based on the applicant’s assessment described above, the staff concluded that under the design-basis tornado wind loads, the structural integrity of the seismic Category I structures will not be compromised from the siding missile strikes in compliance with GDC 2 and GDC 4 in 10 CFR Part 50. Therefore, Open Item OI-SRP3.3.2-SEB1-01 is closed. However, after a tornado strike, the licensee is required to inspect and assess the damage to determine the plant’s operability. If significant damage occurs (such as that described herewith), remedial measures must be taken, including a shutdown. Furthermore, prior to resuming operations, the licensee must demonstrate that no functional impairment remains to those features necessary for continued operation without undue risk to the public health and safety, and that the licensing basis is maintained.

The staff reviewed COL Information Item 3.5-1, including all possible types of missiles generated and the associated kinetic energies produced as a result of external events. The staff determined that, in general, the kinetic energies produced fall within the scope of RG 1.76, “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,” guidelines and thus conform to GDC 4, “Environmental and Dynamic Effects Design Bases,” in Appendix A to

10 CFR Part 50, which requires that SSCs important to safety be protected from the effects of missiles.

3.3.5 Conclusions

There are two major revisions in the DCD Section 3.3. The first change involves the design change of the shield building geometry. The shield building height was reduced by 1.5 m (5 ft). As a result, the total design wind and tornado loads applied on the shield building are altered. The second change involves revision of COL Information Items 3.3-1 and 3.5-1.

The COL Information Item 3.3-1 defines site interface criteria for wind and tornado. Should the site parameters exceed the bounding conditions; the COL applicant will be required to demonstrate that the design conforms to the acceptance criteria.

The COL Information Item 3.5-1 defines acceptable missile type and energy consistent with RG 1.76. The COL applicant is responsible for identifying internal sources and external events. If the missile energy is higher than that depicted in RG 1.76, the effect of an impact must be evaluated to show that it will not impair the structural integrity of the NI safety-related structures. If significant damage occurs (such as that described herein), remedial measures must be taken, including a shutdown. Furthermore, prior to resuming operations, the COL applicant must demonstrate that no functional impairment remains to those features necessary for continued operation without undue risk to the public health and safety, and that the licensing basis is maintained.

The staff reviewed these two proposed changes to the wind and tornado loadings as documented in AP1000 DCD, Revision 16. The staff finds that these two changes do not alter the status of AP1000 wind and tornado loads with regard to meeting the applicable acceptance criteria, including the NUREG-0800 guidelines. The staff also finds that the changes have been properly incorporated into the appropriate sections of the AP1000 DCD, Revision 17. On the basis that the AP1000 wind and tornado loadings continue to meet all applicable acceptance criteria, and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to Section 3.3 of the AP1000 DCD are acceptable.

3.4 External and Internal Flooding

3.4.1 Flood Protection

3.4.1.1 Protection from External Flooding

The proposed changes to the AP1000 DCD adds design features intended to prevent rainfall accumulation on the roofs of the annex, radwaste, and diesel generator buildings, increases the storage volume of one of the fire water tanks and also includes additional features to prevent or limit infiltration of groundwater into seismic Category I structures.

3.4.1.1.1 Evaluation

The staff reviewed all changes related to external flood protection, Section 3.4.1.1.1, in the AP1000 DCD Revision 17, in accordance with NUREG-0800 Section 3.4.2, “Analysis Procedures.” The regulatory basis for this section is documented in NUREG-1793. The staff reviewed the proposed changes to AP1000 DCD Section 3.4.1.1.1, “Protection from External Flooding,” against the applicable acceptance criteria of NUREG-0800 Section 3.4.2.

The staff reviewed the proposed changes to the roof drainage system to determine if it would impact the accumulation of water (ponding) on the roof. The applicant claimed that ponding of water on the roof is still precluded given the additional design features.

In RAI-SRP3.4.1-RHEB-01, the staff asked the applicant to discuss how the addition of parapets with weir openings to the roof drainage system would impact the potential for ponding of water on the roofs of the annex, radwaste and diesel/generator buildings. The applicant's response explained that these buildings are not safety-related seismic Category I structures and that there are no weir openings in the design. The applicant also committed to change the DCD to reflect the change. Given this information and commitment, the staff considers RAI-SRP3.4.1-RHEB-01 to be resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff reviewed the proposed increase in storage volume in the larger firewater storage tank. The amendment seeks to increase the tank volume from 1.514×10^6 to 1.854×10^6 Liters (400,000 to 490,000 gallons).

In RAI-SRP3.4.1-RHEB-02, the staff asked the applicant to assess the impact of the firewater tank failure on safety-related SSCs. The applicants responded in part by referring to DCD Figure 1.2-2. The applicant explained: (1) the distance from the fire water tank to the auxiliary building is 97.54 m (320 ft) and; (2) at that distance the calculated water depth would be 5.59 cm (2.2 in); and (3) that the base of the fire water tank is 30.48 cm (12 in) below the nominal plant grade of 30.48 m (100 ft). The applicant also explained that the site shall be graded with a minimum slope of 1 percent away from the reactor buildings. The applicant also committed to change the DCD to reflect the required site grading. Based upon the depth calculation and the required slope of the site in the vicinity of the tank and NI, along with the commitment to modify the DCD, the staff considers RAI-SRP3.4.1-RHEB-02 to be resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff also reviewed the additional features intended to prevent or limit infiltration of groundwater into seismic Category I structures. These features include embedding piping penetrations into the wall or welding the piping to a steel sleeve embedded in the wall. The amendment also specifies that no access openings or tunnels penetrating the exterior walls of the NI are below grade and that a waterproof membrane or waterproofing system will be installed for the seismic Category I structures below grade.

3.4.1.1.2 Conclusion

The staff reviewed the applicant's proposed changes to the AP1000 external flooding as documented in DCD, Revision 19. The staff finds that the proposed changes in the case of external flooding meet the applicable acceptance criteria defined in NUREG-0800 Section 3.4.1. The staff finds that all of the changes to the AP1000 external flooding are acceptable because they are in compliance with GDC 2 and GDC 4 in Appendix A to 10 CFR Part 50 and 10 CFR 52.63(a)(1)(vii), "Finality of standard design certifications."

3.4.1.2 Internal Flooding

3.4.1.2.1 Summary of Technical Information

In AP1000 DCD, Revision 17, Section 3.4.1.2.2, the applicant proposed the following changes associated with internal flooding to DCD Tier 2 of the certified design:

- The applicant proposed to modify AP1000 DCD Section 3.4.1.2.2.1, “Reactor Coolant System Compartment” to describe that a portion of the steam generator compartment has a low point at 24.38 m (80 ft, 0 in) versus the nominal elevation of 25.30 m (83 ft, 0 in). The basis for this change is described in TR-105, “Building and Structure Configuration, Layout, and General Arrangement Design Updates,” APP-GW-GLN-105, Revision 2, October 2007.

- The applicant proposed to modify AP1000 DCD Section 3.4.1.2.2.1, “Reactor Coolant System Compartment,” to reflect the use of three redundant Class 1E flood-up level indication racks (versus the two originally in the design). The applicant stated that this change was made to assure consistency with DCD Section 6.3.7.4.4.

- The applicant proposed to modify the AP1000 DCD Section 3.4.1.2.2.2, “Auxiliary Building Flooding Events, Level 5 (Elevation 135′-3″)” to remove the discussion of the 568 L (150 gallon) potable water system (PWS) tank rupture in the main mechanical heating, ventilation, and air conditioning (HVAC) equipment rooms, which drains to the turbine building via floor drains or to the annex building via flow under the doors. This change was due to the removal of the PWS from the Westinghouse AP1000 Scope of Certification and the basis for this change is described in TR-124, “Removal of PWS Source and Waste Water System (WWS) Retention Basins from Westinghouse AP1000 Scope Of Certification,” APP-GW-GLN-124, Revision 0, June 2007.

- The applicant proposed the following modifications to AP1000 DCD Section 3.4.1.2.2.2, “PCS Valve Room:”

  (a) The elevation of the PCS Valve Room is changed from 87.33 m (286 ft, 6 in) to 86.82 m (284 ft, 10 in).

  (b) “With the worst crack location being the 6-inch line between the valves and the flow control orifices. This leak is not isolable from the 2.858×10^6 L (755,000 gallon) passive containment cooling system water storage tank above the valve room.”

  (c) “Leakage will flow down to the landing at elevation 277’ 2” where the water will flow through floor drains or under doors to the upper annulus which is then discharged through redundant drains to the storm drain.”

- The applicant proposed to modify AP1000 DCD Section 3.4.1.2.2.3, “Adjacent Structures Flooding Events, Annex Building – Nonradiologically Controlled Areas” to read: “Water accumulation at elevation 100′-0″ is minimized by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area.” This revision eliminates reference to the flow path through the turbine building because the access door at the 30.48 m (100 ft) elevation level was eliminated from the design. The basis for this change is described on page 6 of TR-105.

- The applicant proposed to modify AP1000 DCD Section 3.4.2.2.2.3, “Adjacent Structures Flooding Events, Radwaste Building” to read: “The potential sources of flooding in the radwaste building are the chilled water, hot water, and fire protection systems or from failure of one of the three waste monitor tanks.” The basis for this change is described in TR-116, “Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension,” APP-GW-GLN-116, Revision 0, May 2007.

- The applicant proposed editorial format changes to AP1000 DCD Section 3.4.1.3, “Permanent Dewatering System.” These editorial changes remove references to “COL applicant items.” The basis for this change is discussed in APP-GW-GLR-130, “Editorial Format Changes Related to ‘Combined License applicant’ and ‘Combined License Information Items,’” Revision 0, June 2007. The staff confirmed that these changes are editorial and that no further evaluation is required.

- The applicant also modified Section 4.4, TR-105, to describe structural changes performed to the auxiliary building.

3.4.1.2.2 Evaluation

The staff reviewed all changes related to the internal flooding analysis, Section 3.4.1.2, “Evaluation of Flooding Events,” in the AP1000 DCD, Revision 17, in accordance with NUREG-0800 Section 3.4.1, “Internal Flood Protection for Onsite Equipment Failures.” The staff reviewed the proposed changes to AP1000 DCD Section 3.4.1.2 against the applicable acceptance criteria of NUREG-0800 Section 3.4.1. The following evaluation discusses the results of the staff’s review.

3.4.1.2.2.1 Watertight Doors for Internal Flood Protection

In DCD Section 3.4.1.1.2, the applicant proposed a modification to state that watertight doors, in general, are not needed to protect safe shutdown components from the effects of internal floods with the exception of two watertight doors, those on the two waste holdup tank compartments. In NUREG-1793, Section 3.4.1.2, the staff concluded: “There are no watertight doors used for internal flood protection because they are not needed to protect safe-shutdown components from the effects of internal flooding.”

In its review of DCD Section 3.4.1.1.2, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant’s change. In the DCD, the applicant does not describe those safety components that are protected via the added watertight doors on two waste holdup tank compartments, and does not reference a TR as justification. In RAI-SRP3.4.1-SBPA-01, the staff requested that the applicant demonstrate compliance with GDC 4, by addressing the following:

1) Identify the flood source(s) associated with the spent fuel pit flooding event and the potential flood volume;

2) Provide the volume of a waste hold-up tank compartment; and

3) Identify the safe shutdown components, which are protected by these watertight doors, and provide the design criteria applied for the proper functioning of these doors in the internal flood events considered.

In its July 3, 2008 response, the applicant modified the text of DCD Section 3.4.1.1.2 to reflect that the two watertight doors added during Revision 17 of the DCD were not added to protect safe-shutdown components from the effects of internal floods. These doors were added to provide additional defense-in-depth capability to retain spent fuel pool water within either a single waste holdup tank room or both waste tank rooms to limit consequences of a beyond-design-basis failure of the spent fuel pit. The applicant, in its response, also stated that the volume of a waste hold-up tank compartment is 1.9646×10^5 L (51,900 gallons). Finally, the applicant reiterated that the watertight doors are not used to protect any safe shutdown components. These watertight doors were only added to support the beyond-design-basis accident capability. The applicant stated that the watertight doors were sized to accommodate a water pressure equivalent of 20.73 m (68 ft 0 in) of head, which is conservatively based on the elevation head between the maximum spent fuel pool water level and the finished floor elevation of the tank rooms. No credit is taken for the pool's level being reduced due to the pool volume required to fill the room(s).

On the basis of its evaluation of the revised DCD Section 3.4.1.1.2, the staff finds that the applicant properly identified flood sources associated with the spent fuel pit flooding event, the potential flood volume, the volume of a waste hold-up tank compartment, and the safe shutdown components that are protected by these watertight doors, and the applicant provided an adequate means of protecting safety-related equipment from the identified flood hazards. Therefore, the staff concludes that the applicant's response is acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-01 is resolved.

3.2.1.1.1.1 Building Elevation Changes

In DCD Sections 3.4.1.2.2.1 and 3.4.1.2.2.2, the applicant proposed to make design updates or design description updates to reflect that the steam generator compartment low point elevation is at 24.38 m (80 ft, 0 in) and the passive containment cooling system (PCS) valve room elevation changed from 87.33 m (286 ft, 6 in) to 86.82 m (284 ft, 10 in).

Based on its evaluation of the DCD information, the staff finds that these changes do not affect the existing SER Section 3.4.1.2 assumptions or conclusions related to internal flooding events or protection and are, therefore, acceptable.

3.2.1.1.1.2 Addition of a Redundant Class 1E Flood-Up Level Indication Rack

In DCD Section 3.4.1.2.2.1, the applicant proposed to modify this section to reflect the use of three (versus two) redundant Class 1E flood-up level indication racks. There are no requirements for a specified level of redundancy for these sensors. Moreover, the proposed redundancy level provides an additional layer of protection and, thus, the staff considers that the proposed design demonstrates an increase in reliability when compared to the previously approved design. In addition, the staff notes that this change does not invalidate the evaluation in NUREG-1793 Section 3.4.1.2 because there is no reference to a specific redundancy level, only that redundancy is provided.

Based on its evaluation of the DCD information, the staff concludes that this change does not affect the existing SER Section 3.4.1.2 conclusions related to internal flooding events or protection in the RCS compartment.

3.2.1.1.1.3 Deletion of PWS Tank Rupture in the DCD

In DCD Section 3.4.1.2.2.2, the applicant proposed to delete the discussion of the 0.57 m3 (150 gallons) PWS tank rupture in the main mechanical HVAC equipment rooms that drains to the turbine building via floor drains or to the annex building via flow under the doors. The applicant made this change as a consequence of removing the PWS from the applicant's AP1000 scope of certification. The staff evaluated this change and concludes: 1) this area does not contain equipment whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity; 2) the volume of water supplied by this tank is negligible; and 3) the volume of water from a postulated rupture of this tank or any other flooding source in this area would flow through floor drains to the turbine building or under doors leading to the annex building (which does not contain equipment required to be protected from internal flooding events).

However, since the PWS is no longer included in the scope of the DC, the staff determined that the applicant needed to confirm that this portion of the flooding analysis remains valid, as part of the interface requirements for the site-specific PWS. The staff requested that the applicant address this requirement in RAI-SRP3.4.1-SBPA-06.

In its response to RAI-SRP3.4.1-SBPA-06, the applicant stated that the PWS inside of the standard AP1000 plant is still included in the DCD and the DC and the discussion of the rupture of the 150 gallon PWS tank was inadvertently removed from the DCD. The applicant revised the text in DCD Section 3.4.1.2.2.2 for the potable water tank as follows:

> Water from fire fighting, postulated pipe or potable water storage tank (150 gallons) ruptures in the main mechanical HVAC equipment rooms drains to the turbine building via floor drains or to the annex building via flow under the doors. Therefore, no significant accumulation of water occurs in this room. Floor penetrations are sealed and a 6 inch platform is provided at the elevator and stairwell such that flooding in these rooms does not propagate to levels below.

Based on its evaluation of the revised DCD Section 3.4.1.2.2.2, the staff concludes that the change does not impact the NUREG-1793 Section 3.4.1.2 assumptions, findings, or conclusions related to internal flooding events or protection because the text was revised to match the staff accepted conclusions in DCD Revision 15. On the basis of its review, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-06 is resolved.

3.2.1.1.1.4 Volume of PCS Water Storage Tank

In DCD Section 3.4.1.2.2.2, the applicant corrected the volume of the PCS water storage tank above the valve room to a value of 2.858×10^6 L (755,000 gallons). Although the applicant did not specify the reason for this change, the staff performed its evaluation assuming it is a design change. Given that the proposed volume of water is smaller than the one previously approved, the staff concludes that its effect on the flooding analysis will be conservative.

However, the staff identified areas in which additional information was necessary to complete its evaluation. In NUREG-1793 Section 6.2.1.6, the staff presumed a usable volume of 2.8644×10^6 L (756,700 gallons), which is slightly more, for passive containment heat removal. In RAI-SRP3.4.1-SBPA-02, the staff requested that the applicant clarify and resolve the apparent discrepancy of the volume of water in the PCS water storage tank.

In its response dated July 3, 2008, the applicant stated that it agreed with the staff's conclusion that the AP1000 PCS usable PCS tank volume of 2.8644×10^6 L (756,700 gallons) is appropriate. The indicated value will be corrected in the next version of the DCD. The applicant modified the text to read "…This leak is not isolable from the 756,700 gallon passive containment cooling system water storage tank above the valve room."

Based on its evaluation of the revised DCD Section 3.4.1.1.2 text, the staff finds that the applicant clarified the PCS water storage tank design water volume available either for passive containment cooling or as a potential internal flood source and provided an adequate means of protecting safety-related equipment from the identified flood hazards. On the basis of its review, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-02 is acceptable.

3.2.1.1.1.5 Elimination of flow path through Turbine Building for flooding events in the Annex Building – NRCA

In DCD Section 3.4.2.2.2.3, the applicant eliminated reference to a flow path through the turbine building for flooding events in the annex building, a nonradiologically controlled area (NRCA).

The staff identified areas in which additional information was necessary to complete its evaluation of the applicant's change. In NUREG-1793 Section 3.4.1.2, page 3-21, the staff previously concluded the following:

> The mechanical equipment areas located in the NRCAs include the valve/piping penetration room (Level 3), two main steam isolation valve (MSIV) rooms, and mechanical equipment rooms (Levels 4 and 5). Flood water in these areas is routed to the turbine building or the annex building via drain lines, controlled access ways, or blowout panels which vent from the MSIV room to the turbine building.

In TR-105, the applicant did not justify the effect on the internal flooding analysis results of eliminating the route through the turbine building for flooding events. In RAI-SRP3.4.1-SBPA-03, the staff requested that the applicant clarify the effect of elimination of the turbine building drainage pathway on the internal flooding analysis results.

In its response dated July 3, 2008, the applicant stated that the elimination of the flow path to the turbine building at the 30.40 m (100 ft 0 in) level was compensated by an increase in the egress door opening to Area 4 of the annex building to match the opening previously credited to the turbine building and using the same number of alternate pathways to accommodate the flood source as previously assumed. Therefore, the applicant stated that the flood level has not been changed and remains the same as provided in Revision 15 of the DCD.

The staff identified an area in DCD Section 3.4.2.2.3 in which additional information was necessary to resolve an apparent inconsistency in the paragraph which states:

> The non-Class 1E dc and UPS system (EDS) equipment with regulatory treatment of non-safety-related systems important missions are located on elevation 100’ 0” in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump or flows to the turbine building under doors or to the yard area through doors.

In RAI-SRP3.4.1-SBPA-04, the staff requested that the applicant clarify the apparent discrepancy in the above paragraph. The applicant was requested to clarify whether a drainage path through the turbine building remains in the flood analysis. If there is no longer a drainage path, the applicant was asked to clarify the effect of eliminating this drainage pathway on the results of the internal flooding analysis and to verify that it does not result in any increased water level buildup that would require further evaluation.

In its response dated July 3, 2008, the applicant stated that the paragraph should have been updated consistent with the previous paragraph to reflect the elimination of the flow path to the turbine building at the 30.40 m-0.00 cm (100 ft-0 in) level. The applicant corrected the paragraph in DCD Section 3.4.2.2.3 as follows:

> The class 1E dc and UPS system (EDS) equipment with regulatory treatment of non-safety-related systems important missions is located on elevation 100’-0” in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area.

Based on its evaluation of the responses to RAI-SRP3.4.1-SBPA-03 and RAI-SRP3.4.1-SBPA-04 and the revised DCD Section 3.4.2.2.3 paragraph, the staff finds that the applicant justified that internal flooding analysis results were bounded by the change and provided an adequate means of protecting essential equipment from the identified flood hazards. On the basis of its review, the staff concludes that the applicant’s responses are acceptable and the staff’s concerns described in RAI-SRP3.4.1-SBPA-03 and RAI-SRP3.4.1-SBPA-04 are resolved.

3.2.1.1.1.6 Addition of Three Waste Monitor Tanks to Flooding Analysis

In DCD Section 3.4.1.2.2.3, the applicant included three additional potential sources of flooding, namely: “failure of one of the three waste monitor tanks.” The original design included three 56781 L (15,000 gallons) radwaste monitor tanks which are located in the auxiliary building. In TR-116, the applicant added three additional 56781 L (15,000 gallons) radwaste monitor tanks located in the radwaste building. The additional capacity resulted from evaluation of utility operational needs, and their addition required enlarging the building footprint of the radwaste building.

The staff finds that these changes do not affect the staff conclusions regarding flooding protection requirements in the radwaste building since this building does not house equipment required to be protected from the effects of flooding. Based on its evaluation of the DCD information, the staff concludes that the change does not impact the existing SER Section 3.4.1.2 assumptions, findings, or conclusions related to internal flooding and is acceptable.

3.2.1.1.1.7 Structural Changes Performed to the Auxiliary Building (Change 11)

In TR-105, Section 4.4, the applicant described structural changes performed to the auxiliary building. In RAI-SRP3.4.1-SBPA-05, the staff requested that the applicant clarify if these changes had any impact on the internal flooding analysis. The applicant was requested to confirm that the auxiliary building internal flooding analysis described in DCD Section 3.4.1.2.2.2 was updated to reflect these changes or remained valid. Further, the applicant was asked to discuss how these changes affect the auxiliary building analysis with initiating events in the annex building, given that some of the proposed changes involve additional connections between the annex building and the auxiliary building.

In its response dated July 3, 2008, the applicant stated that changes described in TR-105 Section 4.4 have no impact on the internal flooding analysis as described in DCD Section 3.4.1.2.2.2 and the analysis remains valid. The applicant stated that the structural changes in connections between the annex building and auxiliary building do not have any impact on the auxiliary building flooding analysis with initiating events in the annex building because the connection points are above the elevation of the drainage paths credited for these events.

On the basis of its evaluation, the staff finds that this is a design description update change which does not impact the auxiliary building internal flooding analysis because the revised connection points are above the elevation of the drainage paths credited for these events. Therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.4.1-SBPA-05 is resolved.

3.2.1.1.2 Conclusion

The staff identified acceptance criteria based on the design's meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 2 and GDC 4. The staff reviewed the AP1000 internal flooding design for compliance with these requirements, as referenced in NUREG-0800 Section 3.4.1, and determined that the design of the AP1000 internal flooding is acceptable because the design conforms to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 internal flooding as documented in AP1000 DCD, Revision 17. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 internal flooding to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. On the basis that the AP1000 internal flooding design continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 internal flooding are acceptable.

3.4.2 Analytical and Test Procedures

The AP1000 is designed so that the maximum hydrodynamic water forces considered due to internal flooding, external flooding, and groundwater level changes caused by extreme environmental events do not jeopardize safety of the plant or the ability to achieve and maintain safe shutdown conditions. The analytical procedures for internal flooding are described in Section 3.4.1.2, "Evaluation of Flooding Events," where changes were reviewed with regard to their acceptability. In this subsection, the review will be focused on changes related to external flooding events and their impacts on the structural integrity of the safety related buildings.

3.4.2.1 Summary of Technical Information

With regard to adjacent structures flooding events involving the radwaste building, the proposed change to the DCD adds one more source of potential flooding from failure of one or more of the three added waste monitor tanks in the radwaste building. The basis for this change is described in TR-116.

3.4.2.2 Evaluation

The staff reviewed all changes related to the external flooding analysis, Section 3.4.1.1, “Flood Protection Measures for Seismic Category I Structures, Systems, and Components,” in the AP1000 DCD Revision 16, in accordance with NUREG-0800 Section 3.4.2, “Analysis Procedures.” The regulatory basis for this subsection is documented in NUREG-1793. The staff reviewed the proposed changes to AP1000 DCD Section 3.4.2.2 relevant to external flooding against the applicable acceptance criteria of the NUREG-0800 Section 3.4.2. The review of the internal flooding was described in Section 3.4.1.2, “Internal Flooding.”

The staff reviewed the change with regard to the impact on the hydrodynamic load to determine its acceptability. Since the proposed change adds three additional water tanks of 56781 L (15,000 gallon) capacity each, collapse of the radwaste building (which is a likely scenario) will have a consequence of both internal and external flooding due to the release of a large quantity of liquid from failed tanks. Since all SSCs contained in the building are non-safety related, damage by internal flooding is of no safety concern. Scenarios involving internal flooding are thus acceptable to the staff because of the evaluation contained herein. However, the release of large amounts of water from the three simultaneously failed tanks could result in external flooding to the NI structures important to safety, thereby generating extra hydrodynamic loads to the seismic Category I structures. An analysis showing these additional loads exerted from external flooding will not impair the structural integrity of the safety-related buildings is required. The staff requested that the applicant perform such an analysis in RAI-SRP3.4.2-SEB1-01:

> The design of the radwaste building has been changed to incorporate three new additional liquid waste monitor tanks and the associated piping systems (see TR-116). Provide an analysis to show that external flooding caused by the release of the liquid from tank rupture and collapse of the radwaste building due to safe shutdown earthquake (SSE) or other extreme environmental events will not impair the structural integrity of the adjacent nuclear island (NI) structures.

The applicant responded to RAI-SRP3.4.2-SEB1-01 in a letter dated December 1, 2009. The applicant stated that the increase in flood level would be 15 cm (6 in) more, added to the probable maximum flood (PMF) level due to the collapse of the 3 existing water tanks located in the auxiliary building. However, the associated extra hydrodynamic forces induced were simply stated as insignificant but not evaluated. A quantitative evaluation of the generated hydrodynamic loads showing they are insignificant on the impact to safety is needed to close this open item. This concern was identified as Open Item OI-SRP3.4.2-SEB1-01. In the response of this open item dated June 10, 2010, the applicant provided detailed calculations to arrive at additional water level of 15 cm (6 in), hydrostatic pressure of 1.53 kPa (0.032 kip per square foot (ksf)), and hydrodynamic pressure of 21.6 kPa (0.45 ksf) in APP-1000-CCC-0007, Revision 0. The staff performed an onsite review on the report regarding the methodology, input parameters and calculation procedure, and confirmed the acceptability of the report. The results of the analysis in the report showed that additional water pressures, static as well as dynamic, and increased flood level due to the rupture of water tanks are insignificant on the impact to safety or to impair safety functions needed to be performed by the NI structures. Accordingly, the staff concludes that the change meets the relevant requirements of 10 CFR Parts 50 and 52 and GDC 2 and GDC 4 to Appendix A of 10 CFR Part 50.

The staff reviewed AP1000 DCD Impact Document APP-GW-GLE-012, Revision 0, “Probable Maximum Precipitation Value Increase.” On August 26, 2008 an RAI-SRP2.4-RHEB-01 was presented to the applicant to clarify the maximum groundwater values. This information will affect design basis static and hydrodynamic effective loads applied to seismic Category I structures. This concern was identified as Open Item OI-SRP2.4-RHEB-01 regarding the PFM level and normal groundwater level. In a letter dated September 21, 2009 the response to this open item re-confirms the design-basis PFM at the grade 30.48 m (100 ft) El., and the normal groundwater level up to 29.87 m (98 ft) El. The surface water flooding may prevent outside access to the plant site. The AP1000 is designed to allow isolation for a period of seven days without an increase in safety risk. Thus, the maximum design groundwater elevation is set at 29.87 m (98 ft) El. The staff found that the clarifications in the response to the open item are acceptable and this open item is closed. Accordingly, based on the evaluations described above, the staff concluded that the change does not significantly impact the existing SER Section 2.4 assumptions and conclusions related to changes in ground water levels or protection based on 10 CFR Parts 50 and 52 and associated acceptance criteria GDC 2 and GDC 4 in the Appendix A to 10 CFR Part 50.

3.4.2.3 Conclusions

The staff reviewed the applicant’s proposed changes to the AP1000 external flooding as documented in DCD, Revision 17. The staff finds that the proposed changes in the case of external flooding meet the applicable acceptance criteria defined in the NUREG-0800 Section 3.4.2. The staff also finds that the design changes have been incorporated into the appropriate sections of the AP1000 DCD, Revision 19. Based on the evaluations performed herein, the staff finds that all of the changes to the AP1000 external flooding are acceptable because they are in compliance with the 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

3.5 Missile Protection

3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

3.5.1.4.1 Introduction

GDC 2, in part, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions.

GDC 4, in part, requires that SSCs important to safety shall be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.

With respect to protection of SSCs from missiles generated by tornadoes and extreme winds, the staff reviews the design of nuclear power facilities and considers the design to be in compliance with GDC 2 and GDC 4 if it meets the guidance in RG 1.76, Positions C.1, “Design-Basis Tornado Parameters,” and C.2, “Design-Basis Tornado-Generated Missile Spectrum.”

In RG 1.76, automobile missiles generated by tornadoes are considered to impact at an altitude of less than 9.14 m (30 ft) above plant grade.

The staff reviewed the design of protection of SSCs from missiles generated by tornadoes and extreme winds for an AP1000 facility. In NUREG-1793, the staff concluded that the AP1000 design meets the requirements of GDC 2 and 4 with respect to protection against the effects of natural phenomena such as tornadoes and hurricanes and tornado generated missiles. The design also meets the guidance of RG 1.76 with respect to the identification of missiles generated by natural phenomena. In the initial Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 COL application Final Safety Analyses Report (FSAR) Section 3.5.1.4, “Missiles Generated by Natural Phenomena,” the applicant incorporated by reference Section 3.5.1.4 of the DCD, Revision 16, with one departure that a postulated automobile tornado missile impact is not limited to the height of 9.14 m (30 ft) above grade on the NI. The applicant stated that the effects of a postulated automobile tornado missile impact above the height of 9.14 m (30 ft) above grade on the NI had been evaluated by the applicant.

3.5.1.4.2 Evaluation

During its review of VCSNS COL FSAR Section 3.5.1.4, the staff identified areas in which it needed additional information to complete the evaluation of the departure stated in VCSNS COL FSAR Section 3.5.1.4. Therefore, in an RAI (RAI COL03.05.01.04-1), the staff requested that the applicant describe/provide its evaluation of the postulated automobile tornado missile striking plant structures at elevations higher than 9.14 m (30 ft) above plant grade due to elevated local topography located within 804.67 m (1/2 mile) of the facility. In its responses dated September 10, 2009 and October 21, 2009, the VCSNS applicant discussed TR-133, APP-GW-GLR-133, “Summary of Automobile Tornado Missile 30' above Grade,” Revision 0, dated August 2007. The VCSNS applicant stated that TR-133 envelops the impact analysis of the automobile missile above elevation 39.63 m (130 ft) at VCSNS.

Subsequently, the DCA applicant communicated to the staff that the issue regarding the effects of a postulated tornado generated automobile missile would be addressed generically in the AP1000 DCD rather than in the VCSNS COL FSAR. Accordingly, in its response dated February 16, 2010 to RAI COL03.05.01.04-1, the DCA applicant stated that the postulated tornado-generated automobile missile could impact the plant structures up to the junction of the outer wall of the passive containment cooling water storage tank with the roof of the shield building. The applicant proposed a revision to AP1000 DCD Tier 2 Section 3.5.1.4 to reflect this change and stated that the proposed change, as evaluated in TR-133, would envelop all of the referenced AP1000 sites.

On March 3, 2010, the staff conducted an audit of the automobile tornado missile calculations at the applicant’s Twinbrook office in Rockville, Maryland. The staff issued its audit report on March 24, 2010, which identified nine audit findings. Most of these audit findings were in the nature of requesting clarifications of discrepancies between TR-133 and the DCD and more detailed descriptions regarding the protection provided for the AP1000 facility against tornado generated automobile missiles (i.e., justification for why the passive containment cooling water tank was excluded from the automobile missile, justification for why the y-axis label was blacked out from Figure 1 in APP-GW-GLR-133, justification for why temporary blockage of the air-inlets in the shield building was not a concern, etc.). The most significant area of concern is the evaluation of the global effect of an automobile impact on the shield building including stress.

In addition, during the structural review of TR-133, Revision 0, the staff identified an issue related to the forcing function used in the report as an input for assessing damage due to the automobile impact in the safety analysis, and found that the report did not provide any basis or justification for the input of the forcing function used for the automobile missile impact. To address this concern, the applicant committed to update TR-133 to justify the use of the forcing function. Based on the review, the staff agreed that, because of the similarity of the impact, it is appropriate to use the same forcing function to perform the damage assessment. Accordingly, the applicant committed to add this report as a reference in TR-133, Revision 1. On May 28, 2010, the applicant submitted Revision 1 of TR-133. The staff reviewed TR-133, Revision 1, and confirmed that the forcing function used as a basis for the analysis was added to the report.

Also, in its letter of May 27, 2010, the applicant provided responses to the staff's concerns regarding the evaluation of the global effect of an automobile impact on the shield building, including stress. These staff concerns, the applicant's responses, and the staff's evaluation of the applicant's responses are described below:

In the event of an automobile missile strike on the nuclear island structures 9.14 m (30 ft) above grade, there would be two safety concerns for the seismic Category I structures: (1) local damage; and (2) global damage. The staff reviewed the analysis of local damage in APP-1000-CCC-015, Revision 0 entitled: "Nuclear Island-Tornado Missile Automobile Impact 30' Above Grade." In the report, the applicant considered an impact area 2.01 m by 1.31 m (6.6 ft by 4.3 ft) by the automobile missile with a shear area 0.39 m x 0.60 m (1.29 ft x 1.98 ft) at the weakest location. The shear resistance of the RC wall was assessed at 112.99 pounds per square inch (psi), and the maximum shear stress induced by the impact was calculated to be 89.15 psi. Since the applied shear stress is less than the concrete wall shear resistance, the applicant concluded that the wall is able to resist the impact from being punched through. On this basis, the staff considers that the local damage concern at the impact spot is resolved. Another local damage concern is the crack initiation at the siding missile strike site. If the site is located at a critical section, the crack may grow unstably under the maximum stress induced by the automobile missile impact force as well as the strong tornado wind load. This safety concern was addressed in Section 3.3.4.

In addressing the global damage concern, the applicant provided a safety analysis under Audit Item 8, page 6 of 7 in its Response to RAI COL03.05.01.04-1, Revision 1, dated March 24, 2010. In the report, the possibility of failure at the connector joints of the shield building structure was considered. The analysis showed that an impact force of 3425 kilonewtons (kN) (770 kip) from the automobile missile strike will give rise to a shearing force of 3425 kN (770 kip) and a bending moment of 155.3 meganewton-meter (MN-m) (114,540 kip-ft) at the RC/ steel and concrete composite (SC) connection. The shear resistance at the weakest SS site is 104.1 MN (23,400 kip) and bending moment resistance 3929 MN-m (2,898,000 kip-ft), far exceeding the applied load exerted by the missile. This provided assurance that the connector will not fail under the automobile missile strikes.

The safety concerns of global failure due to sliding and overturning at the base were addressed in the May 13, 2010 audit. The safety analysis was provided in APP-1000-CCC-007, Revision 0 entitled: "Further Evaluation of Potential Tornado Missiles on Nuclear Island." In the report, the resistant shear and bending moment of the building were shown to far exceed the applied shear and bending moment induced by the auto impact with a safety factor of up to 300. However, the review by the staff revealed that the analysis used an incorrect bending moment arm: the center of rotation should be at the base rather than at the connector. The analysis also failed to take the tornado load of 1586 MPa (230 ksi) into account. As a result, the safety factor was dramatically reduced to less than 30 after the corrections. The applicant committed to make the corrections to APP-1000-CCC-007. The staff reviewed APP-1000-CCC-007, Revision 1 and confirmed that the corrections were made.

Based on the safety analysis performed by the applicant against global as well as local failure due to an automobile missile strike 58 m (193 ft) above grade, the staff reviewed and accepted that assurance has been provided that the structural integrity of the NI structures will not be compromised and that the change complies with 10 CFR Part 50 Appendix A, GDC 2 and GDC 4.

In addition, in Enclosure 1 to the letter dated May 27, 2010, the applicant proposed to revise the first bullet under AP1000 DCD Section 3.5.1.4 as follows:

> A massive high-kinetic-energy missile, which deforms on impact. It is assumed to be a 4000-pound automobile impacting the structure at normal incidence with a horizontal velocity of 105 mph or a vertical velocity of 74 mph. This missile is considered at all plant elevations up to 30 feet above grade. In addition, to consider automobiles parked within half a mile of the plant at higher elevations than the plant grade elevation, the evaluation of the automobile missile is considered at all plant elevations up to the junction of the outer wall of the passive containment cooling water storage tank with the roof of the shield building. This elevation is approximately 193 feet above grade. This evaluation bounds sites with automobiles parked within half a mile of the shield building and auxiliary building at elevations up to the equivalent of 163 feet above grade.

Based on its review and audit of the applicant's responses to the above-cited RAI and the applicant's proposed revision to the AP1000 DCD Section 3.5.1.4, the staff finds that the AP1000 design continues to meet the requirements of GDC 2 and GDC 4 with respect to its ability to withstand the effects of natural phenomena and contains plant features that adequately protect against the postulated automobile tornado missile. Therefore, the staff considers its concerns described in RAI-COL03.05.01.04-01 resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.5.1.4.3 Conclusions

The staff reviewed the applicant's proposed changes to the AP1000 postulated tornado automobile missile analysis. The staff finds that the proposed changes related to the postulated tornado-generated automobile missile analysis meet the applicable acceptance criteria defined in NUREG-0800 Section 3.5.1.4. The staff finds that the changes related to postulated tornado automobile missiles are acceptable because they are in compliance with 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.

3.5.3 Barrier Design Procedures

3.5.3.1 Summary of Technical Information

The commitment to address in the combined license information (DCD COL Information Items 3.3-1, "Wind and Tornado Site Interface Criteria" and 3.5-1, "External Missile Protection Requirements"), onsite interface criteria for missile generation and wind and tornado loadings by the COL applicant is met in TR-5, Revision 4. The proposed changes to supply the details of the Information Items are incorporated into the DCD as follows:

- Evaluation of generic wind and tornado loadings on structures,
- Provision of the plant specific site plan and comparison with the typical site plan shown in Figure 1.2-2 of the DCD Section 1.2,
- Discussion of missiles produced by tornadoes and other external events, and
- Evaluation of other buildings for collapse and missile generation.

The staff evaluations are focused on the demonstration that any exceedances or differences in the evaluation results from those specified in the DCD do not compromise the safety of the nuclear power plant.

3.5.3.2 Evaluation

The AP1000 DCD Revision 16, Tier 2, proposed closure of COL Information Items 3.3-1 and 3.5-1 in Section 3.5. In order to close out the COL Information Items, the following items must be addressed by the COL applicant:

(1) Tornado-Initiated Building Collapse (Information Item 3.3-1)

If the COL applicant has adjacent structures different from the typical site plan shown in Figure 1.2-2 of DCD Section 1.2, a justification must be provided to show that they will not collapse or that their failure will not impair the structural integrity of the NI safety-related structures. The structures in the typical site plan have now been evaluated for tornado-initiated failure or collapse. The analysis shows that they will not compromise the safety of the NI structures or result in reclassification of their seismic categories.

The staff reviewed the analysis and found that the procedure followed NUREG-0800 Section 3.5.3, "Barrier Design Procedures," and conformed to applicable codes and RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)." This is acceptable; however, there is one issue that requires further investigation: The radwaste building was evaluated for the potential impact of its collapse on the NI structures to demonstrate that it would not impair the structural integrity of the NI safety-related structures (see DCD Section 3.7.2.8.2). However, because of the addition of three liquid radwaste monitor tanks (see TR-116), which completely alters the structural dynamic characteristics of the building, it is not clear whether this conclusion is still valid. This concern was identified as Open Item OI-SRP3.7.2-SEB1-02. Additional information on this open item is in Section 3.3.4 of this report. As discussed in Section 3.3.4, the safety concern in this open item was that, in the event of a collapse of the radwaste building during a design-basis tornado strike, the three water tanks inside the building were identified as a potential threat to safety if they were allowed to get loose to become a high energy damaging missile. In an attempt to close this open item, the applicant provided a safety analysis in APP-1000-CCC-007, Revision 0 titled, "Further Evaluation of Potential Tornado Missiles on Nuclear Island," showing that during a design-basis tornado event the anchor supports for the three water tanks have adequate resistant strength to prevent the tanks from breaking away to become missiles. On May 13, 2010, the staff performed an onsite review on this TR at the applicant's Twinbrook office. The staff conducted an independent confirmatory analysis and confirmed that so long as the tornado wind speed does not exceed 141.3 m/s (316 mph), the water tanks will not become damaging missiles. Since the design-basis tornado wind speed is set at 134.1 m/s (300 mph) in the DCD, a safety margin of 5 percent is obtained. Detailed reviewed results were discussed in Section 3.3.4. Based on the assurance provided by the TR submitted by the applicant, the staff finds that it is acceptable, and this open item is closed.

(2) Missiles generated by external events (Information Item 3.5-1)

The AP1000 tornado missiles used for design are defined in Table 2.2.-1 of the DCD Section 3.5.1.4 in terms of missile type versus energy spectrum, which is consistent with RG 1.76. Other than by tornado, missiles may also be generated from external events such as transportation accidents or explosions. The COL applicant is responsible for identifying sources in the plant and the external events that could produce missile(s) that threaten the integrity of AP1000 safety-related SSCs. The missile energy should be compared with the table in Section 3.5.1.4. If the external event missile has higher kinetic energy than that given in the table, the effect of the impact must be evaluated to show that it does not compromise the safety of the AP1000 safety-related structures.

The staff reviewed this item, and found that this extra requirement in the barrier design procedure demanded in the Information Item 3.5-1 conforms to the procedure outlined in NUREG-0800 Section 3.5.3 and the criteria dictated by GDC 4 of Appendix A to 10CFR Part 50, which require that SSCs important to safety be protected from the effects of missiles, and GDC 2 concerning the capability of the structures, shields and barriers to protect SSCs important to safety from the effects of natural phenomena. However, there is one remaining issue that requires further evaluation. The issue is related to the missiles that are produced by the potential blow-off of the siding. In the annex building as well as turbine building, metallic insulated siding is permitted to blow off during the extreme environmental event. It appears that the resulting missile in this case does not belong to any missile types listed in Table 2.2-1. Moreover, it is not clear whether the energy spectrum in the table bounds the missile energies associated with the siding-generated missiles.

By letter dated December 23, 2008, the applicant responded to RAI-SRP3.3.2-SEB1-01 regarding the issue of missiles that are produced by the potential blow-off of the siding on the annex building as well as the turbine building. In its response, the applicant indicated that “The automobile in the missile spectrum included in the AP1000 would appear to bound the mass and energy of sheet metal siding. Also there are no safety-related structures, systems, and components outside of the Auxiliary Building and Shield Building. The walls of these buildings are reinforced concrete at least 2 ft thick. Tornado driven siding would not be expected to be a challenge to reinforced concrete walls.” The staff notes that the construction of the shield building is not RC and can best be described as “steel-concrete-steel modular wall construction.” It is likely that the siding missile can penetrate the steel sheet of the modular wall of the shield building and the RC roof. Thus, the reanalysis of the shield building for a tornado-driven siding missile was Open Item OI-SRP3.3.2-SEB1-01. In a letter dated March 24, 2010, the applicant responded to Open Item OI-SRP3.3.2-SEB1-01 regarding the issue of damage induced by siding missiles. In the response, it is concluded that the penetration will be zero according to the basic assumptions, methodology and detailed calculations presented in the TR, APP-1000-CCC-007, Revision 0, “Further Evaluation of Potential Tornado Missiles on Nuclear Island.”

An independent confirmatory analysis performed by the staff showed that for a metallic plank missile, with a mass of 7.8 kg (17.2 lb), flying at a velocity of 134.1 m/s (300 mph), the corner impact on the shield building could cause substantial damage in the form of major cracks several feet long and that a noticeable opening might take place. Details of the analysis are discussed in Section 3.3.4. Nevertheless, because of the large dimensions of the structures, a total collapse of the building is not likely, due to the residual strength of the components (e.g., inner steel panel of the S-C wall or intact rebar in the RC roof). Thus, the structural integrity would still be maintained.

Based on the evaluations described above, the staff concluded that, under the design-basis tornado wind loads, the structural integrity of the seismic Category I structures will not be compromised by the siding missile strikes and that those structures are, thus, in compliance with GDC 2 and GDC 4 in Appendix A to 10 CFR Part 50. However, after a tornado strike, the licensee is required to inspect and assess the damage to determine the plant's operability. If significant damage occurs (such as that described herewith), remedial measures must be taken, including shutdown. Furthermore, prior to resuming operations, the licensee must demonstrate that no functional impairment remains to those features necessary for continued operation without undue risk to the public health and safety, and that the licensing basis is maintained.

3.5.3.3 Conclusions

COL Information Item 3.3-1 defines the design procedure in the case of tornado-initiated building collapse. Should the nonsafety-related building collapse, the COL applicant will be required to demonstrate that the design procedure for the barriers to protect the neighboring Category I structures conforms to the acceptance criteria dictated by NUREG-0800 Section 3.5.3 and GDC 2 and GDC 4 in Appendix A to 10 CFR Part 50.

COL Information Item 3.5-1 defines acceptable missile type and energy consistent with RG 1.76. The applicant is responsible for identifying internal sources and external events that have potential of generating hazardous missiles. If the missile energy is higher than that specified in RG 1.76, the effect of impact must be evaluated as an extra requirement in the barrier design procedure to show that it will not impair the structural integrity of the adjacent NI safety-related structures.

The staff reviewed these two changes in Section 3.5.4, COL Information against the NUREG-0800 guidelines and acceptance criteria regarding the barrier design procedure. Based on the discussion described above by letter dated December 23, 2008, the applicant responded to RAI-SRP3.3.2-SEB1-01 regarding the issue of missiles that are produced by the potential blow-off of the siding on the annex building as well as the turbine building. In its response, the applicant indicated that "The automobile in the missile spectrum included in the AP1000 would appear to bound the mass and energy of sheet metal siding. Also there are no safety-related structures, systems, and components outside of the Auxiliary Building and Shield Building. The walls of these buildings are reinforced concrete at least two feet thick. Tornado driven siding would not be expected to be a challenge to reinforced concrete walls." The staff notes that the construction of the shield building is not RC and can best be described as "steel-concrete-steel modular wall construction." It is likely that the siding missile can penetrate the steel sheet of the modular wall of the shield building and the RC roof. Thus, the reanalysis of the shield building for a tornado-driven siding missile is Open Item OI-SRP3.3.2-SEB1-01. By letter dated March 24, 2010, the applicant responded to Open Item OI-SRP3.3.2-SEB1-01 regarding the damage issue induced by siding missiles. In the response, it is concluded that the penetration will be zero according to the basic assumptions, methodology and detailed calculations presented in the APP-1000-CCC-007, Revision 0, “Further Evaluation of Potential Tornado Missiles on Nuclear Island.”

3.6 Protection against the Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

3.6.1.1 Summary of Technical Information

Section 3.6.1, “Postulated Piping Failures in Fluid Systems Inside and Outside Containment,” of the AP1000 DCD, Revision 15, was approved by staff in the certified design. In the AP1000 DCD, Revision 17, the applicant has proposed to make the following changes to this section of the certified design:

1. In DCD Section 3.6.1.1, “Design Basis,” paragraph J, the applicant proposed to revise those secondary, nonsafety-related components that are used to mitigate postulated line ruptures. The applicant’s justification characterized this change as an editorial change that provides consistency with TR-86, “Alternate Steam and Power Conversion Design,” (APP-GW-GLN-018).

2. In DCD Section 3.6.1.3.3, “Special Protection Considerations,” the applicant proposed to delete the following statement in the criterion for instrumentation required to function following a pipe rupture: “In the event of a high-energy line break outside containment, the only safety-related instrumentation that could be affected is the pressure and flow instrumentation in the MSIV compartment conditions resulting from a 1-square-foot break from either main steam or feedwater line in the MSIV compartment as required in order to perform its safety functions.” The bullet now states that instrumentation required to function following a pipe rupture is protected. The justification for this change is discussed in TR-125, “Corrections to Tier 1 ITAAC 2.2.4 and Tier 2 Section 3.6.1.3.3 and 10.3,” APP-GW-GLR-125, Revision 0, May 2007.

3. In DCD Section 3.6.4.1, “Pipe Break Hazards analysis,” the applicant provided COL actions that reference back to the design basis criteria in Section 3.6.1. The applicant has proposed to revise this COL item to direct the COL applicant to address the completion of the as-designed pipe break hazards analysis.

3.6.1.2 Evaluation

The staff reviewed all changes to the Section 3.6.1 in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.” The regulatory basis for Section 3.6.1 of the AP1000 DCD is documented in NUREG-1793. The staff reviewed the proposed changes to the AP1000 DCD Section 3.6.1 against the applicable acceptance criteria of NUREG-0800 Section 3.6.1. The staff’s review of DCD Section 3.6.1 was limited to postulated piping failures outside containment. The staff’s evaluation of the postulated piping failures inside containment is discussed in Section 3.6.2 of this report.

The following evaluation discusses the results of the staff’s review.

3.6.1.2.1 Design Basis Assumptions

In DCD Revision 16, Tier 2, Section 3.6.1, the applicant provided the design basis and criteria for the analysis needed to demonstrate that safety-related systems are protected from pipe ruptures. This DCD section enumerates the high- and moderate-energy systems, which are potential sources of the dynamic effects associated with pipe ruptures. It also defines separation criteria.

One of the design-basis assumptions used in the dynamic effects analysis for pipe failures included the secondary components (e.g., turbine stop, moisture separator reheater stop, and turbine bypass valves). These valves are credited with mitigating the consequences of a postulated steamline break (given a single active component failure).

In its review of DCD Revision 16, Section 3.6.1, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant's change. In Revision 16 to the DCD Section 3.6.1.1 to paragraph J, the applicant amended the list of secondary components to include the turbine control and stop, the turbine interceptor and reheat stop, and the turbine bypass (steam dump) valves. However, in DCD Section 3.6.1.3.3, the secondary components list consisted of the turbine stop, the moisture separator reheater stop, and the turbine bypass valves, which was inconsistent with paragraph J of DCD Section 3.6.1.1. In RAI-SRP3.6.1-SBPA-01, the staff requested that the applicant resolve the inconsistency identified between Sections 3.6.1.1 and 3.6.1.3.3.

In its response dated July 3, 2008, the applicant acknowledged the inconsistency and confirmed that the non-safety-related valves used to mitigate postulated line ruptures, given the failure of no more than one MSIV, are:

- Turbine Control and Stop Valves
- Turbine Bypass Valves
- Moisture Separator Reheat Supply Steam Control Valves

These valves are identified in the AP1000 Technical Specification (TS) Bases (DCD Section 16.1, B3.7.2), which states that "[t]he non-safety related turbine stop or control valves, in combination with the turbine bypass, and moisture separator reheat supply steam control valves, are assumed as a backup to isolate the steam flow path given a single failure of an MSIV."

In addition, the applicant stated, that based on their review, the inconsistency was not only in Section 3.6.1.1, paragraph J and in Section 3.6.1.3.3 of the DCD, but also in Section 10.3.1.1 of the DCD.

As part of its response, the applicant provided a markup of the AP1000 DCD, Revision 16, Sections 3.6.1.1, 3.6.1.3.3, and 10.3.1.1 to rectify the inconsistencies. The staff has confirmed that the AP1000 DCD, Revision 17 has included these changes.

On the basis of its review and evaluation, the staff finds that the revisions to the DCD have corrected the inconsistencies in the application; therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.6.1-SBPA-01 is resolved.

3.6.1.2.2 Protection Mechanisms

In DCD Revision 16, Tier 2, Section 3.6.1, the applicant provided the measures used in the AP1000 design to protect safety-related equipment from the dynamic effects of pipe failures. These measures include physical separation of systems and components, barriers, equipment shields, and pipe whip restraints. The specific method used depends on objectives such as adequate allowance for equipment accessibility and maintenance.

Separation between redundant safety systems is the preferred method used to protect against the dynamic effects of pipe failures. Separation is achieved using the following design features:

- locating safety-related systems away from high-energy piping
- locating redundant safety systems in separate compartments
- enclosing specific components to ensure protection and redundancy
- providing drainage systems for flood control.

The staff identified an area in which additional information was necessary to complete its evaluation of the applicant's change. There was an inconsistency between TR-125 and the DCD revision that needed to be resolved. In DCD Revision 16, Section 3.6.1.3.3, the applicant provided specific protection considerations and provided the justification for revising the DCD. However, in TR-125, the applicant deleted the entire second bullet, while in Revision 16 to the DCD, the first sentence of the second bullet remained (e.g., "Instrumentation required to function following a pipe rupture is protected.") In RAI-SRP3.6.1-SBPA-02, the staff requested that the applicant resolve this inconsistency.

In its response dated July 3, 2008, the applicant stated that in developing the markup for TR-125, Revision 0, the entire second bullet of DCD Section 3.6.1.3.3 as reflected in Section 5 of TR-125, was erroneously deleted. When preparing the DCD text, however, the first sentence of the second bullet was correctly retained since it is applicable to all safety-related instrumentation located in a harsh environment.

The applicant further stated that TR-125 Section 5.0 will be revised to be consistent with DCD Section 3.6.1.3.3 Revision 16.

On the basis of its review and evaluation, the staff finds that the change to the second bullet in DCD Section 3.6.1.3.3, Revision 19, is accurate with respect to the design specifications. The proposed change ensures that all safety-related instrumentation in a harsh environment is protected from the consequences of a pipe break. Therefore, the staff finds the applicant's response to be acceptable and the staff's concern described in RAI-SRP3.6.1-SBPA-02 is resolved.

3.6.1.2.3 COL Actions

In DCD Revision 17, Section 3.6.4.1, the applicant modified COL actions with respect to pipe break hazard analysis to address the completion of the as-designed pipe hazards analysis report. While this COL information item does not change the design basis criteria as discussed in Section 3.6.1, the modified COL Information confirms that the piping design meets the criteria provided in Section 3.6.1.3.2 (AP1000 DCD, Table 1.8-2, COL Information Item 3.6-1). The staff evaluation of the modified COL Information Item is contained in Section 3.6.2 of this report. The staff finds that the changes to the AP1000 DCD Section 3.6.4.1 COL action are acceptable, as they relate to the protection of safety related components outside containment from the effects of a pipe break. The protection of safety related components inside containment, from the effects of a pipe break, is discussed in Section 3.6.2 of this report.

3.6.1.3 Conclusions

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the applicant's application for design certification met the requirements of Subpart B to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of the AP1000 DCD, Section 3.6.1, "Postulated Piping Failures in Fluid Systems Inside and Outside Containment," the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 2 and GDC 4. The staff reviewed the AP1000 postulated piping failures in fluid systems outside containment design for compliance with these requirements, as referenced in NUREG-0800 Section 3.6.1 and determined that the design of the AP1000 postulated piping failures, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 protection of safety related component inside containment as documented in AP1000 DCD, Revision 19. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 postulated piping failures in fluid systems outside containment to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 19. On the basis that the AP1000 postulated piping failures in fluid systems outside containment design continue to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 DCD Section 3.6.1 are acceptable.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2.1 Summary of Technical Information

AP1000 DCD Section 3.6.4.1 identifies a COL Information Item 3.6-1. The original Combined License Information Item commitment was:

> Combined License applicants referencing the AP1000 certified design will complete the final pipe whip restraint design and address as-built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. The as-built pipe rupture hazards analysis will be documented in an as-built Pipe Rupture Hazards Analysis Report.

Subsequent to the issuance of NUREG-1793, in a letter dated January 14, 2008, APP-GW-GLR-134 through Revision 4 and AP1000 DCD Revisions 16 and 17, the applicant made some DCD changes related to COL Information Item 3.6-1.

3.6.2.2 Evaluation

The staff's review of the changes made to COL Information Item 3.6-1 are based on the pertinent information included in DCD Revisions 16 and 17, TR-6, "AP1000 As-Built COL Information Items," APP-GW-GLR-021; APP-GW-GLR-074, "Pipe Break Hazards Analysis"; and APP-GW-GLR-134, "AP1000 DCD Impacts to Support COLA Standardization," through Revision 4 as well as the proposed DCD Revision 17 changes included in the applicant's letter dated January 14, 2008, and December 5, 2008. In APP-GW-GLR-021 and APP-GW-GLR-074, the applicant proposed to modify the COL information item and provided a pipe rupture hazards analysis report for staff's review. The applicant stated that the report addressed and documented, on a generic basis, design activities required to complete COL Information Item in DCD Section 3.6.4.1 in the AP 1000 DCD. The applicant further stated that when the NRC review of APP-GW-GLR-074 is complete, the included activities to address the COL information item in Section 3.6.4.1 will be considered complete for COL applicants referencing the AP1000 DC. On the basis of its review of that report, the staff found that there were numerous areas in the report that were incomplete (e.g., ASME Code Class 1 piping fatigue evaluation, the complete design of the jet shields and pipe whip restraints, use of seismic response spectrum, etc.). The staff therefore, determined that the pipe rupture analysis documented in APP-GW-GLR-074 could not be considered complete and the proposed revision to the COL Information Item 3.6-1 concerning the COL applicant's responsibility was not acceptable.

Subsequently, in a letter dated January 14, 2008, the applicant proposed to revise AP1000 DCD Revision 16, Section 3.6.4.1 to address the staff's comments on the completeness of APP-GW-GLR-074. Based on its review of the information included in DCD Revisions 16 and 17, the staff determined that the following additional information concerning the acceptability of the proposed COL holder item is needed:

1a. The staff maintains that the pipe rupture hazards analysis report in APP-GW-GLR-074 is incomplete. 10 CFR 52.79(d)(3), "Contents of applications; technical information in final safety analysis report," and RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR [Light-Water Reactor] Edition)," Section C.III.4.3 allows the applicant to propose an alternative to the COL information item that cannot be resolved completely before the issuance of a license. It requires the applicant to provide sufficient information to justify why that item cannot be completed before the issuance of a license. Furthermore, it states that the applicant should provide sufficient information on this item to support the NRC licensing decision and also to propose a method for ensuring the final closure of the item including implementation schedules to allow the coordination of activities with the NRC construction inspection program following issuance of the COL. The current DCD and APP-GW-GLR-134 do not cover the level of detail described in 10 CFR 52.79(d)(3) and RG 1.206, Section C.III.4.3. The applicant is requested to propose an alternative along with the described justification including implementation schedules to allow the coordination of activities with the NRC construction inspection program.

1b. In some of the DCD Tier 1 tables of the system based design description and ITAAC, the applicant includes an acceptance criterion, which states that for the as-built piping, a pipe rupture hazards analysis report exists and concludes that protection from the dynamic effects of a line break is provided. It should be noted that the pipe rupture hazards analysis report is required for all the piping systems (with the exception of leak-before-break (LBB) piping) that are within the scope of NUREG-0800 Section 3.6.2. The staff's concern is that the current AP1000 system based ITAAC tables do not reflect that. The applicant is requested to address how the system based ITAAC approach addresses all the piping systems which are within the scope of NUREG-0800 Section 3.6.2 and are required to be included in a pipe rupture hazards analysis performed in accordance with the criteria outlined in DCD Sections 3.6.1.3.2 and 3.6.2.5.

2. In DCD Revision 16, Section 3.6.2.5 under high energy break locations, the applicant stated that for ASME Class 1 piping terminal end locations are determined from the piping isometric drawings. Intermediate break locations depend on the ASME Code stress report fatigue analysis results. These results are not available at design certification. For the design of the AP1000, breaks are postulated at locations typically associated with a high cumulative fatigue usage factor. The applicant further stated that these locations are part of the as-built reconciliation as discussed in Section 3.6.4.1. As discussed in this RAI question 1a, the determination of break locations is a part of the as-designed pipe rupture hazards analysis and is not part of the as-built reconciliation. The applicant is requested to address this concern and to revise the DCD Section 3.6.2.5 accordingly.

In a letter dated December 5, 2008, the applicant provided its response to the above RAIs. Based on its review of the applicant's response, the staff agreed with the applicant that the as-built reconciliation of the pipe rupture hazards analysis report is included in the ITAAC tables of the DCD which was previously reviewed and found acceptable by the staff. However, with respect to the as-designed pipe rupture hazards analysis, the staff found that the applicant has not yet adequately addressed the staff's concern relating to the completion of the as-designed pipe rupture hazards analysis report issue. Specifically, it is not clear that the as-designed pipe rupture hazards analysis report will include all piping systems within the scope of NUREG-0800 Section 3.6.2 and the report will include all the information as outlined in AP1000 DCD Sections 3.6.1.3.2 and 3.6.2.5. Moreover, it did not clearly address the process including the milestone for the completion of the as-designed pipe rupture hazard analysis report for all piping systems within the scope of NUREG-0800 Section 3.6.2. Furthermore, based on the review of the RAI response provided by some AP1000 COL applicants, the staff found that there is a difference of opinion between the applicant and the COL applicants as to what will be completed and, at this point, the design is not adequately addressed.

On April 9, 2009, the staff, in an AP1000 Design Centered Working Group meeting, conveyed these specific concerns to the applicant and AP1000 COL applicants. Subsequently, the applicant requested a meeting with the staff to discuss its plan, schedule and scope of the as-designed pipe rupture hazard analysis report. The meeting was held on May 20, 2009, at the applicant's Twinbrook office. During the meeting, the applicant indicated that it would complete an as-designed pipe rupture hazard analysis in accordance with the criteria outlined in DCD Sections 3.6.1.3.2 and 3.6.2.5 for all the piping systems within the scope of NUREG-0800 Sections 3.6.1 and 3.6.2 by the end of 2009 with the exception of the completion of the design for some pipe whip restraints. The remaining pipe whip restraint design would be completed by COL applicants referencing the AP1000 certified design. In addition, the applicant indicated that it would include all the above information in an RAI response to address the staff's concerns related to the as-designed piping rupture hazard evaluation issue. In response to the applicant's proposed approach, the staff indicated that it is important that all the representative AP1000 pipe whip restraint designs be completed by the applicant in its as-designed pipe rupture hazards analysis report. Also, the applicant was requested to include a discussion in its RAI response to explain what pipe whip restraints design will be completed to support staff's audit and how they are representative of the ones that will be used in the AP1000 design.

By letters dated June 30 and July 22, 2009, the applicant provided its response to RAI-SRP3.6.2-EMB2-01 R3, RAI-SRP3.6.4-EMB2-01 R3, and RAI-SRP3.6.2-EMB2-01 R4, respectively. Based on its review of these RAI responses, the staff found that the applicant had not clearly and adequately addressed all the issues discussed in the May 20, 2009, meeting and, for some areas, the information included in these RAI responses was different from what the applicant stated in that meeting.

In its response to RAI-SRP3.6.2-EMB2-01 R4, the applicant stated that the as-designed pipe rupture hazards analysis report, with the exception of some pipe whip restraint and jet shield designs, would be completed by December 31, 2009, and that some pipe whip restraint and jet shield designs were not expected to be completed in time to support the advanced SER with no open items. Completion of the remaining pipe whip restraint and jet shield designs will require a modified COL information item to be addressed in the COL applications. The applicant further indicated that portions of the evaluation to complete the COL Information Item might be completed during the COL application review or after the license was issued. It should be noted that during the May 20, 2009, meeting, the applicant indicated that to support the staff's audit, it would complete an as-designed pipe rupture hazard analysis in accordance with the criteria outlined in DCD Sections 3.6.1.3.2 and 3.6.2.5 for all the piping systems (including nonsafety-related piping systems, were not addressed in the applicant's RAI responses) within the scope of NUREG-0800 Sections 3.6.1 and 3.6.2, with the exception of the completion of the design for some pipe whip restraints (as opposed to pipe whip restraints and jet shields indicated in the applicant's RAI responses). Furthermore, based on the information included in the RAI responses, it was not clear what pipe whip restraints and jet shields design would be completed by December 31, 2009, and how they are representative of the ones that would be used in the AP1000 design. The applicant was, therefore, requested again to describe in detail which pipe whip restraint and jet shield designs would be completed to support staff's audit and how these completed pipe whip restraints and jet shield designs are representative of for the AP1000 design.

In its response to RAI-SRP3.6.2-EMB2-01 R4, the applicant also proposed some changes to DCD Sections 3.6.2.5 and 3.6.4.1. The proposed changes did not make clear that the effects of leakage and through-wall cracks in both high and moderate energy pipes (as opposed to moderate energy pipes identified in the RAI response) are to be evaluated as part of the as-designed pipe rupture hazards analysis. It should be noted that both dynamic effects and environmental effects resulting from breaks/leakage cracks need to be evaluated for high energy pipes, while only environmental effects resulting from leakage cracks need to be evaluated for moderate energy pipes. Moreover, based on the review of the proposed DCD Section 3.6.4.1 changes, it appeared that the final completion of all pipe whip restraint and jet shield designs is a COL information item; however, it was not clearly labeled as one. The applicant was requested to clearly identify it as a COL information item or to make it an ITAAC item. This item was considered as Open Item OI-SRP3.6.2-EMB2-01.

In its response to Open Item OI-SRP3.6.2-EMB2-01, the applicant submitted a letter dated April 16, 2010. The applicant proposed that the full scope of the as-designed pipe rupture hazards analysis be addressed in COL Information Item 3.6-1. The revised COL Information Item 3.6-1 would state that COL applicants referencing the AP1000 design would complete the as-designed pipe rupture hazards analysis according to the criteria outlined in DCD Sections 3.6.1.3.2 and 3.6.2.5. SSCs identified (in DCD Tier 2, Table 3.6-3) to be essential targets protected by associated mitigation features would be confirmed as part of the evaluation, and updated information would be provided as appropriate. The pipe whip restraint and jet shield design included the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design would be completed prior to installation of the piping and connected components. The COL Information Item 3.6-1 would be addressed by the COL applicant in a manner that complies with NRC guidance provided in RG 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52,” and outlined in Appendix 14.3A of the DCD. The applicant further stated that the applicant would continue to work towards completion of the as-designed pipe rupture hazards analysis, and would submit a licensing topical report to the staff documenting completion of the effort and referencing the applicable design documents. The report would support the closure of the COL Information Item for the reference standard plant.

In addition, in its response to Open Item OI-SRP3.6.2-EMB2-01, the applicant also revised DCD Tier 1, Table 3.3-6 Line Item 8, which requires an as-built reconciliation of the pipe rupture hazards analysis be completed prior to fuel load. The as-built reconciliation of the pipe rupture hazards analysis is to conclude that systems, structures and components identified as essential targets are protected from dynamic and environmental effects of postulated pipe ruptures.

Based on its evaluation of the above information, the staff determines that the applicant’s response adequately addressed the staff’s concerns described in Open Item OI-SRP3.6.2-EMB2-01. Specifically, the proposed COL Information Item 3.6-1 and the guidance outlined in Appendix 14.3A of the DCD will ensure that the COL applicants referencing the AP1000 design will complete the as-designed pipe rupture hazards analysis report and will make it available for staff’s verification in accordance with the guidance outlined in Appendix 14.3A of the DCD. In addition, the as-designed pipe rupture hazards analysis will be performed for all the piping systems within the scope of NUREG-0800 Sections 3.6.1 and 3.6.2 in accordance with the criteria outlined in DCD Sections 3.6.1.3.2 and 3.6.2.5. Therefore, the applicant’s RAI response adequately addressed all the staff’s safety questions/concerns identified in Open Item OI-SRP3.6.2-EMB2-01. In addition, the revised DCD Tier 1, Table 3.3-6 Line Item 8, provides an acceptable as-built reconciliation of pipe rupture hazards analysis and will ensure that systems, structures and components identified as essential targets are protected from dynamic and environmental effects of postulated pipe ruptures. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.6.2.3 Conclusion

The staff concludes that the applicant’s proposed changes to the COL information item are acceptable because they meet the applicable 10 CFR Part 52 requirements. Specifically, the applicant has provided an acceptable alternative along with the technical justification as described in 10 CFR 52.79(d)(3) and RG 1.206 Section C.III.4.3 regarding COL information items that cannot be resolved before the issuance of a license.

3.6.3 Leak-Before-Break

3.6.3.1 Introduction

In Revision 16 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 3.6-2 by addressing the as-designed LBB evaluation in TR-8, “AP1000 Leak-Before-Break Evaluation of As-Designed Piping,” APP-GW-GLR-022, Revision 1. COL Information Item 3.6-2 in the DCD, which is also discussed in NUREG-1793, as COL Action Item 3.6.3.1-2, specifies requirements for the as-designed evaluation of LBB characteristics in AP1000 LBB piping systems. The applicant submitted TR-8 for the staff’s review to demonstrate that it has met the requirements of COL Information Item 3.6-2. In Revision 15 to the AP1000 DCD, Section 3.6.4.2 states:

> Combined License applicants referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves documented in Appendix 3B. The Combined License applicant may perform leak-before-break evaluation for a specific location and loading for cases not covered by the bounding analysis curves. Successfully satisfying the bounding analysis curve limits in Appendix 3B may necessitate lowering the detection limit for unidentified leakage in containment from 0.5 gpm to 0.25 gpm. If so, the Combined License holder shall provide a leak detection system capable of detecting a 0.25 gpm leak within 1 hour and shall modify appropriate portions of the DCD including subsections 5.2.5, 3.6.3.3, 11.2.4.1, Technical Specification 3.4.7 (and Bases), Technical Specification Bases B3.4.9, and Technical Specification 3.7.8 (and Bases). The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

In Revision 16 of the AP1000 DCD, the applicant proposed to resolve COL Information Item 3.6-2 by addressing the as-designed LBB evaluation in TR-8. The revision to Section 3.6.4.2 of the DCD states:

> The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-022, and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant.
>
> The following words represent the original Combined License Information item commitment, which has been addressed as discussed above:
>
> Combined License applications referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves [BACs] documented in Appendix 3B. The Combined License applicant may perform leak-before-break evaluation for a specific location and loading for cases not covered by the bounding analysis curves. Successfully satisfying the bounding analysis curve limits in Appendix 3B may necessitate lowering the detection limit for unidentified leakage in containment from 1.9 L/m (0.5 gpm) to 0.9 L/m (0.25 gpm). If so, the Combined License holder shall provide a leak detection system capable of detecting a 0.9 L/m (0.25 gpm) leak within 1 hour and shall modify appropriate portions of the DCD including subsections 5.2.5, 3.6.3.3, 11.2.4.1, Technical Specification 3.4.7 (and Bases), Technical Specification Bases B3.4.9, and Technical Specification 3.7.8 (and Bases). The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

The scope of this evaluation does not include piping stress analysis reports whose outputs are used as inputs to this LBB evaluation.

In Revision 16 to the AP1000 DCD, the applicant proposed to delete COL Information Item 3.6-3 for the LBB evaluation. COL Information Item 3.6-3 in the applicant DCD, which is also discussed in NUREG-1793 as COL Action Item 3.6.3.1-1, specifies requirements for the as-built evaluation of LBB characteristics in certain AP1000 piping systems. The applicant submitted APP-GW-GLR-021, Revision 0, for staff review to demonstrate that COL Information

Item 3.6-3 may be deleted. In Revision 15, Section 3.6.4.3 to the AP1000 DCD, COL Information Item 3.6-3 states:

> Combined License applications referencing the AP1000 certified design will address: 1) verification that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping excluded from consideration of the dynamic effects of pipe break are bounded by the leak-before-break bounding analysis; 2) a review of the Certified Material Test Reports or Certifications from the Material Manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied; and 3) complete the leak-before-break evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B. The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

In Revision 16 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 3.6-3 by deleting the text in Section 3.6.4.3. The applicant provided TR-6 as justification to delete COL Information Item 3.6-3.

In Revision 17 of the AP1000 DCD, the applicant proposed to change the composition of the main steam line (MSL) piping material. Previously, in Table 3B-1 of the DCD (Revision 15), the applicant identified the MSL material to be utilized as ASME SA-333 Grade 6. In Revision 17 of the DCD, the applicant revised its DCD in Section 3.6.3 and Appendix 3B to reflect the use of ASME SA-335 Grade 11 Alloy steel. The applicant stated that the composition of the main steam lines was revised to minimize the potential for erosion-corrosion.

3.6.3.2 Evaluation

3.6.3.2.1 COL Information Item 3.6-2

GDC 4 of Appendix A to 10 CFR Part 50 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects. The staff reviewed changes related to this section as it relates to the LBB analysis.

The applicant has designated TR-08 to be the “LBB Evaluation Report,” as referenced in the COL information item. This report has reproduced, with limited modifications due to minor piping design changes, DCD BACs documented in Appendix 3B for the AP1000 LBB piping subsystems. For each AP1000 LBB piping subsystem, there is, however, extra information added to the BAC in TR-08: a point showing the normal stress (the horizontal axis) and the maximum stress (the vertical axis) based on the piping stress analysis report for the system. The normal stress is defined as the stress at the critical location of a AP1000 LBB piping subsystem due to normal loads (deadweight + pressure + thermal expansion), which are combined by the algebraic sum method. The maximum stress is defined as the stress at the critical location of a AP1000 LBB piping subsystem due to maximum loads (deadweight + pressure + thermal expansion + safe shutdown earthquake/inertia + safe shutdown earthquake/anchor motion), which are combined by the absolute sum method. The objective of this review is to verify that the stress pair (the normal stress and the maximum stress) for each AP1000 LBB subsystem has been calculated appropriately by the applicant based on the piping stress report results.

An RAI was issued on August 29, 2006. A revision for one of the RAI questions was issued on September 11, 2006. RAI-TR08-001 is related to the revised BAC for the 20.3 cm (8 in) automatic depressurization system (ADS) Stages 2 and 3 (upper tier) piping. RAI-TR08-002 is related to the LBB evaluation process which starts with the piping stress report results and ends with the stress pairs for all the AP1000 LBB piping subsystems. RAI-TR08-003 is related to a design change to remove the reducing tee and to add a 35.6 cm x 20.3 cm (14 in x 8 in) reducer in the upper tier of the ADS piping. The applicant provided responses to the staff RAIs in a letter dated September 29, 2006. Since quantitative information was provided for the revised BAC requested in RAI-TR08-001, this RAI is resolved. In RAI-TR08-003 the staff requested that the applicant confirm the piping design changes and their effect on the corresponding BACs. In its September 29, 2006, response, the applicant clarified the specific changes made to the piping design and confirmed that the changes do not require additional BACs because the BACs for 15.2 cm, 20.3 cm, and 35.6 cm (6 in, 8 in, and 14 in) piping were developed for the ADS upper tier piping, and are, thus, bounding. Therefore, RAI-TR08-003 is resolved.

RAI-TR08-002 requested additional information regarding the process of calculating the stress pair for each AP1000 LBB piping subsystem based on the corresponding piping stress report results. This involved computer software examinations, LBB calculation demonstrations, and on-site documents review. Consequently, an audit was conducted on August 29 and 30, 2006. During the audit, the staff examined line by line two post processing software designed by different applicant subcontractors for LBB evaluations. In addition, the staff audited the LBB stress-pair calculations for one software application using an as-designed AP1000 ADS upper-tier piping and calculations for another software application using a sample passive core cooling (PXS) piping system. As a result of this audit, the staff found that the two post-processing software applications result in accurate stress pairs for the LBB evaluation, and the use of the software procedure, which does not rely on manual input of technical data, would minimize human error.

The staff's evaluation was based on the piping stress analysis results using seismic loadings associated with an AP1000 plant situated on a hard-rock (HR) site. At that time, the applicant was considering revising the AP1000 seismic design to include plants situated on soil sites as well. Because the seismic loadings for a plant situated on a soil site are likely to be higher than those for a plant situated on a HR site, the LBB analyses for AP1000 plants situated on soil sites (or other sites other than HR) would likely be affected. Thus, the staff's evaluation of the LBB analyses considered seismic loadings for HR sites only. The staff confirmed that each added stress point is enveloped by the BAC curve of its piping system, indicating that all piping systems have met the requirements of COL Information Item 3.6-2. Hence, the applicant has demonstrated that all as-designed AP1000 LBB subsystems for plants situated on HR sites meet the GDC 4 requirements for LBB applications so that the dynamic effects of postulated high-energy line pipe breaks need not be evaluated for these systems.

In addition, the proposed justification for eliminating COL Information Item 3.6-2 is based on the staff's review of the applicant's detailed design information that demonstrates that the LBB calculations are bounded by the bounding analysis curves in the AP1000 DCD. The LBB as-designed analyses as described in TR-08 (APP-GW-GLR-022) are applicable to all COL applications referencing an AP1000 plant situated on a HR site. The final as-built LBB analyses will be verified by the staff as part of its verification of ITAAC.

TR-08 also confirmed that the leak detection capability limit for unidentified leakage inside containment is 1.9 Lpm (0.5 gpm) as described in the DCD.

By letter dated June 20, 2008, the applicant addressed the LBB evaluation for AP1000 plants situated on other-than-HR sites as follows:

> The other-than-hard-rock site seismic spectra are included in the piping analysis that is within the piping DAC review. The LBB evaluation results will indicate that the bounding analysis curves for piping that was evaluated for the other-than-hard-rock seismic input are acceptable and can be addressed as part of the piping DAC review.

The staff reviewed the applicant's response to address LBB for as-designed piping using other than HR site seismic spectra. The applicant stated that for plants situated on other-than-HR-sites, the as-designed LBB analyses would be completed in conjunction with piping design acceptance criteria (DAC), now a COL item (see Section 3.6.2). The staff will review the final as-built LBB analyses results as part of its review of the COL item to verify that the LBB acceptance criteria are met. On the basis of its review of APP-GW-GLR-022 (TR-08), the staff finds that the LBB analysis in TR-08 meets the requirements of GDC 4 and is acceptable; COL Information Item 3.6-2 is closed.

3.6.3.2.2 COL Information Item 3.6-3

GDC 4 of Appendix A to 10 CFR Part 50 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects. The staff reviewed changes related to this section as it relates to the LBB analysis.

TR-06 states that the as-built evaluation of LBB characteristics will be completed after construction of the associated piping systems, as required by the ITAACs, and deletion of the COL Information Item, which requires completion of the as-built evaluation, does not alter the as-designed LBB evaluation. Since the applicant's justification did not address all three requirements in COL Information Item 3.6-3, the staff requested, in a letter dated August 29, 2006, that the applicant justify the proposed deletion of this COL information item in accordance with the following RAI (RAI-TR06-002):

> On page 4 of the report, you propose to delete COL Information Item 3.6-3 regarding the as-built evaluation of leak-before-break piping systems. COL Information Item 3.6-3 has three elements: "1) verification that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping are bounded by the leak-before-break bounding analysis; 2) a review of the Certified Material Test Reports or Certifications from the Material Manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied; and 3) complete the leak-before-break evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B." Report APP-GW-GLR-022 addressed only the third requirement in COL Information Item 3.6-3, and the ITAAC regarding LBB piping systems does not specifically address the first and the second requirements. Please justify your proposed deletion of this COL Information Item by explaining how the first and second requirements (Elements 1 and 2 above) are addressed by your phrase "several ITAAC items."

The applicant's response, dated September 27, 2006, to RAI-TR06-002 states that the relevant ITAACs that specify the requirements for LBB evaluations are located in the DCD as Item 6 in

Table 2.1.2-4 for the RCS, Item 6 in Table 2.2.3-4 for the passive core cooling system, Item 6 in Table 2.2.4-4 for the steam generator system, and Item 6 in Table 2.3.6-4 for the normal residual heat removal systems. The following is the ITAAC requirement on LBB for these systems:

> 6. Each of the as-built lines identified in Table x.x.x-x as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

Except for the referenced component table number, the ITAAC requirements regarding LBB evaluation are identical for all systems mentioned above. Since the above standard ITAAC requirement regarding an LBB system is not specific enough, it might not be interpreted as including the activities specified in Items 1 and 2 of COL Information Item 3.6-3 if this COL information item were deleted. To relieve this concern, the applicant modified its technical justification for TR-06 by adding the following statement in its September 27, 2006 response:

> The activities that require procurement or fabrication include verification of the stresses, diameter, wall thickness, material, welding process, pressure, and temperature of the as-built piping. The activities that require procurement or fabrication also include a review of the Certified Material Test Reports or Certifications from the material manufacturer to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied.

The above statement in TR-06 is essentially a restatement of the first and second requirements in COL Information Item 3.6-3. The third requirement requires applicants to complete the LBB evaluation by comparing the results of the final piping stress analysis with the bounding analysis curves documented in Appendix 3B of the AP1000 DCD. To address this, a separate report, TR-08, APP-GW-GLR-022, Revision 1, dated July 2006, was submitted by the applicant and provides an evaluation for every as-designed LBB piping. The staff has completed its evaluation of TR-08 in Section 3.6.3.1 of this supplement and finds it acceptable. Although TR-08 significantly simplifies the work related to meeting the ITAAC LBB requirements, it is not meant to replace the ITAAC activity related to LBB. When the as-built piping information becomes available after the COL phase, a final LBB evaluation will be performed by the staff in accordance with the ITAAC scope.

Therefore, the staff found that the DCD changes, as proposed by the applicant in TR-06, meet the requirements of GDC 4 and are acceptable. COL Information Item 3.6-3 is resolved.

3.6.3.2.3 Composition of MSL Material

GDC 4 of Appendix A to 10 CFR Part 50 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects. GDC 4 allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of postulated pipe ruptures when the analyses demonstrate that the probability of pipe rupture is extremely low. The staff reviewed the DCD Revision 17 changes in Section 3.6.3 and Appendix 3B as they relate to affecting the LBB methodology and analysis results.

The identification of SA-335 Grade 11 Alloy material for the MSL is a change from the certified design (Revision 15 of the DCD), which identified the MSL material in Table 3B-1 as SA-333 Grade 6. The applicant stated that SA-335 Grade 11 was selected for the MSL material to minimize the potential for erosion-corrosion. This material contains 1-1/4 percent Chromium that is sufficient to preclude erosion-corrosion degradation in the MSL located inside containment. The staff also reviewed Appendix 3B and Figure 3B-4 in Revision 17 in which the applicant revised its LBB analysis for this material, provided a revised bounding analysis curve for the MSL, and verified that the LBB analysis for this material remained bounding for the AP1000 DCD. On this basis, the staff finds the changes to the DCD associated with the use of SA-335 Grade 11 Alloy material for the MSL to be acceptable.

3.6.3.3 Conclusion

On the basis of its review of the AP1000 report APP-GW-GLR-02 (TR-06), the staff finds that the proposed deletion of COL Information Item 3.6-3 meets the requirements of GDC 4 and is acceptable based on the following: (1) the first two requirements in COL Information Item 3.6-3 are preserved in TR-06, and (2) the third requirement is maintained by meeting ITAAC requirements, as described in Item 6 of Table 2.1.2-4 for the RCS, Item 6 of Table 2.2.3-4 for the passive core cooling system, Item 6 of Table 2.2.4-4 for the steam generator system, and Item 6 of Table 2.3.6-4 for the normal residual heat removal systems. Furthermore, the staff finds that the TR-06 conclusions regarding LBB characteristics in certain AP1000 piping systems are generic and are expected to apply to all COL applications referencing the AP1000 design certification. Therefore, COL Information Item 3.6-3 is deleted.

On the basis of its review of the changes in Revision 17 of the AP1000 DCD, the staff finds that the LBB analysis meets the requirements of GDC 4 and is acceptable.

3.7 Seismic Design

The staff has conducted a detailed technical review of the seismic design and analysis of the AP1000 structures, as documented in AP1000 DCD, Revision 19 and the TRs discussed below. The staff used the guidance provided in Sections 3.7.1, 3.7.2, and 3.7.3 of NUREG-0800 to conduct its review.

In September 2004, the staff issued NUREG-1793 for the AP1000 DCD, Revision 15. In Section 3.7 of NUREG-1793, the staff concluded that the AP1000 seismic Category 1 structures located on the NI were capable of withstanding the AP1000 generic SSE ground response spectra. The SSE (now referred to as the CSDRS) is based on RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 1, for a peak ground acceleration (PGA) of 0.3g. An additional control point at 25 Hz is included as a modification to the RG 1.60 ground response spectral shape. The current AP1000 design certification is applicable only to HR sites. An HR site is defined as having a shear wave velocity (Vs) of the supporting media ≥ 2438.4 m/s (8,000 fps). The staff also concluded that the in-structure response spectra (ISRS) were developed in accordance with staff-accepted methods described in NUREG-0800 Sections 3.7.1 and 3.7.2; and that the applicant had identified and/or implemented analytical methods for seismic system analysis and seismic subsystem analysis, consistent with NUREG-0800 Sections 3.7.2 and 3.7.3.

Subsequent to the issuance of NUREG-1793, the applicant submitted Revisions 16 through 19 to the AP1000 DCD. The applicant also submitted the following TRs:

(1) APP-GW-S2R-010, AP1000 Standard (STD) COL TR-03, “Extension of Nuclear Island Seismic Analyses to Soil Sites,” Revisions 0 through 5. The contents of this report are summarized in the new AP1000 DCD Appendix 3G, “Nuclear Island Seismic Analyses.”

(2) APP-GW-GLR-115, AP1000 STD COL TR-115, “Effect of High Frequency Seismic Content on Structures, Systems, and Components,” Revisions 0 through 3. The contents of this report are summarized in the new AP1000 DCD Appendix 3I, “Evaluation for High Frequency Seismic Input.”

The AP1000 seismic design changes introduced in the revised AP1000 DCD and supporting TRs are discussed in the following paragraphs:

1. Extension to soil sites

   The AP1000 DCD, Revision 15 only addresses the seismic design of AP1000 for an HR site. The AP1000 certified seismic design response spectra (CSDRS) for an HR site are RG 1.60 spectra anchored at 0.3g PGA, with an additional control point specified at 25 Hertz (Hz). The same CSDRS are specified in the AP1000 DCD, Revisions 16 through 19, in which the applicant introduced soil-structure interaction (SSI) analysis to evaluate the seismic response for a range of site conditions, from firm rock (FR) to soft soil (SS). For the original HR case, the applicant applies the seismic design input at the foundation El. 18.3 m (60 ft); for the FR to SS cases, the applicant applies the seismic design input at the finished grade in the free field (El. 30.5 m (100 ft)). The applicant evaluated the structures and developed the ISRS using the enveloped response of the multiple analyses. To support the technical basis for the extension of the AP1000 design to FR and soil sites, the applicant submitted TR-03, and summarized the report in AP1000 DCD Appendix 3G. The staff’s detailed evaluation of AP1000 DCD Appendix 3G and TR-03 is described in Section 3.7.2 of this report.

2. Use of 3-D finite element shell models

   In the AP1000 DCD, Revision 15, the applicant used three dimensional (3D) lumped mass stick models to represent the auxiliary building, containment internal structures (CISs), shield building, and steel containment. In the AP1000 DCD, Revisions 16 through 19, the applicant uses 3D finite element shell models for all NI buildings, except the steel containment. These models are used for the SSI and fixed-base seismic analyses. The detailed descriptions of the models and results of the new analyses are provided in TR-03, and summarized in AP1000 DCD Appendix 3G. The staff’s detailed evaluation of these models is described in Section 3.7.2 of this report.

3. Effect of High Frequency Ground Motion

   The seismic analysis and design of the AP1000 plant is based on the CSDRS, which have dominant energy content in the low frequency range (2-10 Hz). However, recent probabilistic hazard-based, site-specific spectral shapes for the Central and Eastern United States (CEUS) show significant amplification above 10 Hz. This high-frequency amplification exceeds the RG 1.60 spectral amplification upon which the AP1000 CSDRS is based. The applicant has determined that for several candidate CEUS rock sites, the site-specific ground motion response spectra (GMRS) show significant increased amplitude in the high frequency range, which exceeds the CSDRS for the AP1000. The applicant has defined generic AP1000 hard rock high frequency (HRHF) spectra, which exceed the CSDRS above 15 Hz in the horizontal direction and above 20 Hz in the vertical direction. To address the exceedances, the applicant has performed an evaluation to demonstrate that, in general, the high frequency ground motion represents a lower seismic demand on AP1000 SSCs than the CSDRS.

The applicant compared the responses for a sample of SSCs, using both the CSDRS and the HRHF response spectra as seismic inputs. The evaluation included building structures, RPV internals, primary component supports, primary loop nozzles, piping, and electro-mechanical equipment. The applicant's evaluation of HRHF ground motion is described in TR-115, and briefly summarized in the new AP1000 DCD Appendix 3I. The staff's review of the applicant's evaluation of high frequency effects is described in Section 3.7.2 of this report.

4. Application of Incoherency Effects

   The incoherency of seismic waves has been recognized for several decades as having an effect on structures with large dimensions, separate supports, or large distances between supports (e.g., bridges). Until recently, data to support analytical models were scarce. Luco, Abrahamson, Zerva, and others, using data from surface recordings from dense arrays located in Taiwan, Japan, and California, developed coherency models to characterize local variations in free-field ground motions to analytically capture these incoherent effects sustained by structural foundations. These data were previously based on recordings at soil sites. Recently, Abrahamson (2006) extended these coherency models to include the effects at rock sites. This coherency function approximates the known changes of motion based on spatial separation and frequency and has been incorporated into several SSI analysis codes.

   The incoherency of seismic waves generally results in a reduction of structural translational responses when compared with coherent seismic motion, especially in higher frequency ranges (e.g., frequencies greater than 10 Hz). For structures of large dimensions typical of nuclear power plants designs, these translational modes can be reduced due to wave scattering, but torsion and rocking modes can be induced that can result in increased response at locations remote from the center-of-mass.

   The applicant has used seismic motion incoherency in its evaluation of HRHF ground motion effects on AP1000 SSCs. The staff issued DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications," in May 2008, identifying an acceptable approach to consider the effects of incoherency on the NI foundation, specifically for HRHF seismic ground motion. The staff accepted the seismic ground motion coherency function as described in an EPRI report entitled, "Hard-Rock Coherency Functions Based on the Pinyon Flat Array Data," dated July 5, 2007. The applicant indicated that its evaluation is consistent with the staff's ISG. Because this is a first-time implementation of the staff's ISG, the staff conducted independent confirmatory analysis. The staff's detailed evaluation of the applicant's use of incoherency is described in Section 3.7.2 of this report.

3.7.1 Seismic Input

NUREG-0800 Section 3.7.1, "Seismic Design Parameters," provides guidelines for the staff to use in reviewing issues related to the development of seismic input ground motions, percentage of critical damping values, and supporting media for seismic Category I structures. The following evaluation addresses the proposed changes to the seismic design, as described in the amendment to the AP1000 DC. As such, this evaluation revises and supplements the evaluation in corresponding sections of NUREG-1793.

3.7.1.1 Design Ground Response Spectra

In AP1000 DCD Tier 1, Section 5.0, the applicant described the AP1000 CSDRS. The staff verified that the AP1000 CSDRS remain unchanged from the AP1000 DCD, Revision 15. In AP1000 DCD Tier 2, Section 3.7.1.1, the applicant indicated that the AP1000 CSDRS have been established with a PGA of 0.3g for the AP1000 design, in both the horizontal and vertical directions. The design response spectra are based on RG 1.60 with an additional control point specified at 25 Hz. The spectral amplitude at 25 Hz is 30 percent higher than the RG 1.60 spectral amplitude.

In AP1000 DCD, Tier 2, Section 2.5.2, the applicant provided a description of how the AP1000 CSDRS are compared to the site-specific GMRS. The CSDRS are compared to the site-specific GMRS at different locations depending on the site characteristics. In AP1000 DCD Section 3.7.1.1, the applicant states that the CSDRS are applied at the foundation level (El. 18.44 m (60 ft 6 in)) in the free field at HR sites and at the finished grade (El. 30.48 m (100 ft)) in the free field at FR and soil sites. Applying the design response spectra at the foundation level in the free field for the HR sites was accepted by the staff during its AP1000 DCD, Revision 15 review. With respect to the FR and soil sites, the staff finds that the applicant's approach of applying the design response spectra at the surface (in the free field) for both FR and soil sites is acceptable, because it is in accordance with the guidance described in NUREG-0800 Section 3.7.1.

The staff noted, however, that AP1000 DCD Section 3.7.1, Revision 17, did not provide a basis for satisfying 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," which requires the horizontal component of the SSE ground motion in the free field at the foundation elevation to have a PGA of at least 0.1g and an appropriate response spectrum. To address this concern, the staff issued RAI-SRP3.7.1-SEB1-18, requesting the applicant to provide free field in-column response spectra and associated PGA generated for each of the generic-site columns (FR and soil sites) considered. This was identified as Open Item OI-SRP3.7.1-SEB1-18 in the SER with open items.

In a letter dated May 14, 2010, the applicant provided the in-column response spectra at the basemat elevation for each of the generic sites, in Figure RAI-SRP3.7.1-SEB1-18-1, attached to the response. The horizontal PGA at the basemat elevation is above 0.1g for all generic sites. On this basis, the staff determined that the requirements of 10 CFR Part 50, Appendix S, are satisfied; therefore, RAI-SRP3.7.1-SEB1-18 and the associated open item are resolved.

3.7.1.2 Critical Damping Values

In AP1000 DCD, Tier 2, Section 3.7.1.3, the applicant described the critical damping values assigned to seismic Category I SSCs. The staff reviewed the critical damping values specified for seismic analysis of Category I SSCs, and noted that the applicant made no changes to the critical damping values in AP1000 DCD Section 3.7.1.3, between Revision 15 and Revision 17. However, the staff has updated the NUREG-0800 Section 3.7.1 guidance on critical damping, to reference Revision 1 of RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Both documents were issued in March 2007. RG 1.61, Revision 1, now addresses response-compatible structural damping, electrical distribution system damping (e.g., cable trays), and electrical component damping (e.g., cabinets, panels). The staff noted that the applicant's specified damping values were higher than the RG 1.61, Revision 1, values in these areas.

The staff issued RAI-SRP3.7.1-SEB1-16, requesting the applicant to specify whether it planned to use the RG 1.61, Revision 1, damping values; or to provide the technical basis for concluding that the damping values the applicant is using will provide sufficient conservatism. In a letter dated May 14, 2009, the applicant submitted its response for each area questioned by the staff:

Response-Compatible Structural Damping

The applicant stated that the HRHF ISRS generated from the analysis are used in evaluating the acceptability of safety-related equipment and components that might be susceptible to HRHF seismic excitation. Acceptability of the equipment is demonstrated by performing an HRHF ISRS seismic test run, after seismic testing to the AP1000 CSDRS ISRS.

In order to address the possibility that the HRHF ISRS may have been underestimated, the applicant included an additional seismic test margin of approximately 30 percent in the HRHF seismic screening evaluation of safety-related equipment vulnerable to HRHF excitation. This is accomplished by using the 3 percent damping HRHF ISRS in place of the 5 percent damping HRHF ISRS as the required response spectra (RRS) for testing. This approach compensates for the increase in structural response that would have been predicted if the HRHF seismic structural analysis had used 4 percent structural damping instead of 7 percent structural damping.

The staff determined that the 30 percent increase in the RRS is sufficient to compensate for the potential under-prediction of structural response, and is acceptable to meet the intent of the guidance in RG 1.61, Revision 1 (i.e., to use response-compatible structural damping when developing ISRS).

Cable Tray Damping

The applicant stated that the AP1000 design for cable tray support configurations uses construction (Unistrut with bolted connections) covered by the Systematic Evaluation Program (SEP) test program (conducted by ANCO Engineers Inc.). Based on observations during the tests, the high damping values within the cable tray system are provided mainly by the movement, sliding, or bouncing of the cables within the tray. The applicant also stated that the limiting condition for design of the AP1000 standard cable tray supports is for full cable tray weight. The damping value being used for the design of this condition is 10 percent, which is consistent with the value listed in AP1000 DCD Table 3.7.1-1 for full cable trays and related supports. The staff noted that seismic design of full cable trays using 10 percent damping is consistent with the guidance in RG 1.61, Revision 1, and is acceptable.

Electrical Cabinet and Panel Damping

The applicant stated that electrical cabinets and panels employed in safety-related applications are an assembly of structures, subassemblies, and individual components. The electrical cabinets and panels are generally constructed of carbon steel framing members, angle support channels, and panels with a combination of bolted and welded connections designed to support subassemblies and components mounted within. The structural damping of cabinets and panels is a function of the materials, design, mass distribution, and method of interconnection (bolted/welded).

The applicant noted that RG 1.61, Revision 0, defines SSE level damping values as 4 percent for welded steel structures and 7 percent for bolted steel structures; and it is reasonable to perform the analysis of combined bolted and welded structures using an average of the structural damping associated with the bolted or welded steel structures as defined in RG 1.61, Revision 0. In Section 3.7, Table 3.7.1-1 of the AP1000 DCD, Revision 17, the applicant specifies 5 percent damping for electrical cabinets and panels.

The applicant further stated that dynamic structural finite element analyses employ models validated through the use of qualification test program results. The response of the finite element method (FEM) is developed and validated against test data and used as the basis for any modifications that are needed. The results of seismic testing are used in the correlation of dynamic in-equipment response, and the modal and structural damping results from the resonant search test data are used to determine the natural frequency of vibration and associated structural damping used in model correlation process. In most instances, this leads to the use of 4 percent and 5 percent critical damping in the finite element analysis.

The staff concluded that, although the RG 1.61, Revision 1, guidance is 3 percent damping for electrical cabinets and panels at the SSE analysis level, the applicant has provided an acceptable technical basis for use of higher damping values. For FEM analyses, damping values of 4 to 5 percent are validated by test results. For static coefficient analyses, the use of 5 percent damping is acceptable, when used in conjunction with a 1.5 multiplier on the spectral peak. Although the 1.5 multiplier is intended to provide margin when a multidegree of freedom system or component is analyzed by the static coefficient method, in the case of electrical cabinets and panels, the response is single-mode dominant; the 1.5 multiplier on the 5 percent damping spectral peak would compensate for the difference between 3 percent damping and 5 percent damping.

Based on the applicant’s responses and the staff’s evaluation, the response to RAI-SRP3.7.1-SEB1-16 is considered acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Shield Building Structural Damping

In the AP1000 DCD, Revision 17, the applicant changed the design of the shield building from RC construction (7 percent SSE damping in AP1000 DCD Table 3.7.1-1) to steel and concrete composite (SC) -filled module construction (5 percent SSE damping in AP1000 DCD Table 3.7.1-1). The staff issued RAI-SRP3.7.1-SEB1-19, part (a), requesting the applicant to define the damping value(s) used for the SC module walls, and to describe how this value is assigned in the ANSYS and SASSI models.

The staff also noted that the applicant reduced the shield building concrete modulus (Ec) to 80 percent of nominal value, to account for concrete cracking. The 80 percent value is recommended by the Federal Emergency Management Agency (FEMA) when there is minimal load-induced cracking. Since the 80 percent factor is associated with minimal cracking, the staff noted that use of reduced damping may be appropriate, because damping has been recognized as being a function of the structural response level. At low response levels, lower effective viscous damping has been observed; at high response levels, higher effective viscous damping has been observed. In RAI-SRP3.7.1-SEB1-19, part (b), the staff requested that the applicant submit the technical basis for the damping values assumed. This was identified as Open Item OI-SRP3.7.1-SEB1-19 in the SER with open items.

In its response dated August 26, 2010, the applicant stated that 5 percent structural damping was assumed for the SC modules, including the shield building wall, and 7 percent structural damping was assumed for RC structures. The applicant also stated that these damping values were defined in ANSYS and SASSI as a material property defined for each element.

To demonstrate that the assumed damping values for SC and RC are appropriate, the applicant relied on the results of a nonlinear time-history analysis using the ABAQUS finite element code. In this analysis, concrete was allowed to crack in tension. In Figures RAI-SRP3.7.1-SEB1-19-06 through RAI-SRP3.7.1-SEB1-19-09 of the response, the applicant provided plots of maximum principal stress versus time in the SC, and showed that the predicted stresses either were close to, or reached, the tensile cracking limit of 2.06 MPa (43 ksf) during the progress of the analyzed SSE event. The applicant stated that the use of 5 percent damping was justified if element stresses approached this limit. The applicant also provided a contour plot of maximum principal stresses in the shield building, in Figure RAI-SRP3.7.1-SEB1-19-14 of the response. The applicant stated that the results, at 11.33 seconds, indicate cracking in most of the west side of the shield building wall. Similar contour plots for the RC auxiliary building were provided in Figures RAI-SRP3.7.1-SEB1-19-15 through RAI-SRP3.7.1-SEB1-19-17 of the response, at 7.22 seconds, 8.34 seconds, and 10.28 seconds, respectively. The staff's review of these figures identified that stresses reach the RC tensile cracking limit 1.72 MPa (36 ksf) in large expanses of the auxiliary building during the SSE event. Based on the applicant's calculations, indicating tensile cracking of concrete for significant portions of the AP1000 NI, the staff finds the applicant's use of SSE-level damping values of 5 percent for the shield building SC wall and 7 percent for RC to be acceptable. Therefore, Open Item OI-SRP3.7.1-SEB1-19 is resolved.

In its August 26, 2010 response, the applicant also addressed the use of concrete stiffness reduction in linear analysis, to account for the effect of concrete cracking. To demonstrate that using a reduced concrete modulus of 0.8 x Ec in the design-basis seismic analysis of the NI is appropriate to account for stiffness reduction due to concrete cracking, the applicant performed nonlinear ABAQUS analysis, using a smeared concrete cracking model, and compared the results to the results of a linear ABAQUS analysis, which assumed 0.8 x Ec for the concrete modulus. The applicant submitted additional details of this comparison in its response to related Open Item OI-SRP3.8.3-SEB1-03.

The applicant compared the ABAQUS results (linear and nonlinear) to linear ANSYS NI20 results, in order to validate that the ABAQUS models are dynamically similar to the ANSYS design-basis model. The applicant presented response spectra comparisons, in three orthogonal directions, at the shield building roof in Figures RAI-SRP3.7.1-SEB1-19-11 through RAI-SRP3.7.1-SEB1-19-13 of the response. The comparisons show that the nonlinear ABAQUS model results are very similar to and are enveloped by the linear model results, which assume 0.8 x Ec. The applicant also provided a plot of stress-strain for a highly stressed element in the shield building (West wall location), in Figure RAI-SRP3.7.1-SEB1-19-02 of the response. The applicant stated that while principal stress values are at or near the assumed cracking threshold 2.06 MPa (43 ksf), the concrete strains are relatively small; and further stated that the associated secant stiffness would be close to 0.8 x Ec, as shown in Figure RAI-SRP3.7.1-19-01 of the response.

The staff reviewed the applicant's analysis results presented in the response to this open item and in the response to OI-SRP3.8.3-SEB1-03, and determined that the applicant has provided a sufficient technical basis for using a reduced concrete modulus of 0.8 x Ec, to account for stiffness reduction due to cracking. The response is acceptable on the basis that the applicant's comparison of linear (0.8 x Ec) and nonlinear (concrete cracking model) analysis results showed a very good correlation, with the linear model being conservative.

3.7.1.3 Supporting Media for Seismic Category I Structures

In AP1000 DCD, Appendix 3G and accompanying TR-03, the applicant described the supporting media, which define the characteristics of the material providing support for the AP1000 NI. The AP1000 DCD, Revision 15 was certified for supporting media consisting of HR. In the AP1000 DCD, Revisions 16 through 19, the applicant included a range of FR to SS profiles. For each rock/soil profile, the applicant performed SSI analysis in order to demonstrate the seismic adequacy of the AP1000 plant for the range of soil and rock sites. For the design of seismic Category I structures, a set of six design soil profiles of various Vs values were established from parametric studies, as described in AP1000 DCD Appendix 3G and TR-03. The applicant stated that these six profiles are sufficient to envelop sites where the Vs of the supporting medium at the foundation level exceed 304.8 m/s (1000 fps). The design soil profiles include an HR site, an FR site, a soft rock (SR) site, an upper bound soft-to-medium (UBSM) soil site, a soft-to-medium (SM) soil site, and an SS site. The Vs profiles and related governing parameters of the six sites are:

- Hard-rock site - an upper bound case for rock sites using a Vs of 2438.4 m/s (8000 fps).

- Firm-rock site - a Vs of 1066 m/s (3500 fps) to a depth of 36.7 m (120 ft) and base rock at the depth of 36.7 m (120 ft).

- Soft-rock site - a Vs of 731.5 m/s (2400 fps) at the ground surface, increasing linearly to 975.4 m/s (3200 fps) at a depth of 73.12 m (240 ft), and base rock at the depth of 36.7 m (120 ft).

- Upper bound soft-to-medium soil site - a Vs of 430.9 m/s (1414 fps) at ground surface, increasing parabolically to 1034.45 m/s (3394 fps) at 73.2 m (240 ft), base rock at the depth of 36.7 m (120 ft), and ground water at grade level. The initial soil shear modulus profile is twice that of the SM soil site.

- Soft-to-medium soil site - a Vs of 304.8 m/s (1000 fps) at ground surface, increasing parabolically to 731.5 m/s (2400 fps) at 73.15 m (240 ft), base rock at the depth of 36.7 m (120 ft), and ground water is assumed at grade level.

- Soft-soil site - a Vs of 304.8 m/s (1000 fps) at ground surface, increasing linearly to 365.8 m/s (1200 fps) at 73.2 m (240 ft), base rock at the depth of 36.7 m (120 ft), and ground water is assumed at grade level.

The staff reviewed the range of soil profiles and properties identified in AP1000 DCD Revision 17, Section 3.7.1.4, and the iterated Vs profiles presented in Table 3.7.1-4 and Figure 3.7.1-17. In TR-03, Section 4.4, the applicant stated that the range of soil profiles and properties are based on a survey of 22 commercial nuclear power plant sites in the United States. The applicant's survey included sites with Vs ranging from 304.8 m/s (1,000 fps) (SS) to 2438.4 m/s (8,000 fps) (HR). Based on its review, the staff concluded that the applicant has selected a suitable range of site profiles for extending the AP1000 seismic design basis.

3.7.1.4 Conclusion

The staff concludes that Revision 19 to the AP1000 DCD continues to support the seismic design parameters, seismic system analysis, and seismic subsystem analysis for Category I SSCs to meet NRC regulations applicable to the AP1000 DC. The application to amend the AP1000 certified design provides sufficient information to satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," for the seismic design and analysis aspects for Category I SSCs to be used in the AP1000 reactor.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.7.2 Seismic System Analysis

NUREG-0800 Section 3.7.2, "Seismic System Analysis," provides guidelines for the staff to use in reviewing issues related to seismic system analysis. The AP1000 DCD, Revisions 16 through 19, introduced the following significant changes related to AP1000 DCD Section 3.7.2: (1) the applicant performed SSI analysis using the SASSI computer code to extend the AP1000 certified seismic HR design basis to include a range of soil and rock sites; (2) the applicant used 3D shell models instead of 3D stick models for performing dynamic analysis of the NI; (3) the applicant evaluated the effects of HRHF ground motion on the design of AP1000 SSCs; and (4) the applicant used a seismic wave incoherency model in the HRHF analysis, to reduce the effective demand.

The applicant's technical discussion of these changes is incorporated in several sections of the AP1000 DCD and the applicable TRs. The applicant added AP1000 DCD Appendix 3G to document the extension of the seismic design basis to a wide range of soil and rock sites. AP1000 DCD Appendix 3G summarizes the content of TR-03. The applicant also added AP1000 DCD Appendix 3I to briefly summarize the HRHF analysis documented in TR-115. The staff's evaluations of TR-03 and TR-115 are included in Section 3.7.2 of this SER.

The applicant also moved most of the analysis details previously in AP1000 DCD, Revision 15, Section 3.7.2, to the new AP1000 DCD Appendix 3G. The building stick models used in the original HR DC analyses, described in the AP1000 DCD, Revision 15, have been replaced by 3D shell FEMs for the SSI analyses (using SASSI) and for the updated fixed-base analyses (using ANSYS). In addition, the equivalent static acceleration methodology, described in the

AP1000 DCD, Revision 15 for the detailed design of the buildings, has been replaced by response spectrum analysis (RSA) for the auxiliary/shield building (ASB) and for the CISs.

The applicant's use of a seismic wave incoherency model to effectively reduce HRHF ground motion represents the first application of the ISG-1 on this subject. As a result, the staff performed an independent confirmatory analysis using the applicant's NI20 SASSI model and NI10/NI20 ANSYS models. The purpose of the staff's confirmatory analysis was to: (1) evaluate the adequacy of NI20 model for seismic analysis of soil sites and the representative HRHF site; (2) verify the correct implementation of an incoherency model; (3) assess the adequacy of the structures sample set selected by the applicant for HF analysis; and (4) assess overall compliance with ISG-1. The results of the staff's confirmatory analysis effort are described in Section 3.7.2.3.4.2 of this SER.

3.7.2.1 Seismic Analysis Methods

In AP1000 DCD, Revision 17, Section 3.7.2.1, the applicant describes the methods used for performing seismic analyses. The applicant stated that the seismic analyses of the NI are performed in conformance with the criteria in NUREG-0800 Section 3.7.2. RSA, the equivalent static acceleration method, the mode superposition time-history method, and the complex frequency response analysis method are performed for the SSE to determine the seismic force distribution for use in the design of the NI structures, and to develop in-structure seismic responses (accelerations, displacements, and floor response spectra [FRS]) for use in the analysis and design of seismic subsystems. In TR-03, Table 4.2.4-1, the applicant provided a summary of the models and analysis methods used by the applicant in the seismic analyses.

The staff reviewed AP1000 DCD Section 3.7.2.1, and related information in Appendices 3G and 3I, and determined that the applicant's seismic analysis methods are not completely consistent with the latest staff guidance in NUREG-0800 Section 3.7.2, Revision 3 (March 2007). This is discussed in detail in Section 3.7.2.7 of this SER.

The applicant accounted for the effects of SSI by using the SASSI analysis code and used 3D models that accounted for the effects of torsional, rocking and translational responses. The staff finds the SASSI analysis code acceptable for performing SSI analysis because it has been independently benchmarked to standard problems for this type of analysis.

As part of the review of the applicant's SSI analysis methods, the staff performed independent confirmatory analysis using FEMs provided by the applicant. As a result of this effort, the staff identified several modeling errors made by the applicant. The staff's confirmatory analysis is described in Section 3.7.2.4.2 of this SER.

3.7.2.2 Natural Frequencies and Responses

In AP1000 DCD, Revision 17, Section 3.7.2.2, the applicant stated that modal analyses are performed for the shell and lumped-mass stick models of the seismic Category I structures on the NI, as described in Appendix 3G.

The staff reviewed the applicant's seismic analyses models described in AP1000 DCD Section 3.7.2.2, Appendix 3G, and TR-03. The staff issued RAI-TR03-32 and RAI-SRP3.7.1-SEB1-06, requesting the applicant to demonstrate the capability of the NI20 and NI10 models to accurately predict all natural frequencies up to the 33 Hz for the AP1000

CSDRS and up to 50 Hz for the HRHF evaluation spectra. The staff's evaluation for these RAIs is in Section 3.7.2.4 of this report.

3.7.2.3 Procedures Used for Analytical Modeling

The staff reviewed AP1000 DCD, Revision 17, Section 3.7.2.3, and related information in Appendix 3G. The staff also reviewed TR-03, which provides the detailed information supporting Appendix 3G. In AP1000 DCD Section 3.7.2.3, the applicant indicated that 3D finite element shell models were developed for the coupled shield and auxiliary buildings, and for the CIS. An axisymmetric finite element shell model of the steel containment vessel (SCV) was also developed. These models provide the basis for the development of the dynamic model of the NI structures. In the dynamic model, the SCV is represented by a lumped mass stick model with properties developed from the SCV axisymmetric model. A separate detailed 3D finite element model of the shield building roof was also developed for detailed design.

The applicant stated that the models of the coupled shield and auxiliary buildings and the CIS are based on the gross concrete section, with the modulus of elasticity reduced to 0.8 times the nominal value, to consider the effect of cracking.

The applicant further stated that seismic subsystems coupled to the overall dynamic model of the NI include the reactor coolant loop model coupled to the CIS model, and the polar crane model coupled to the SCV model. The criteria used for decoupling seismic subsystems from the NI model are taken from Section II.3.b of NUREG-0800 Section 3.7.2, Revision 2.

In TR-03, Section 1.0, the applicant identified the information included in TR-03, in order to update the seismic design basis for AP1000: (1) description of the new 3D shell finite element ANSYS and SASSI models; (2) minor structural changes that are significant; (3) the seismic analysis results for a specified range of soil sites; (4) revised envelope ISRS at six reference locations; and (5) the effect of extending the seismic design basis on the seismic design of the NI structures. The staff noted that the only structural change described in TR-03 was the pressurizer compartment redesign. Therefore, in RAI-TR03-001, the staff requested that the applicant describe the other "minor structural changes that are significant" and explain why the changes to the AP1000 design are necessary.

In its response dated January 18, 2007, the applicant stated that the seismic analysis models, NI10 and NI20, have been revised from those reviewed during the HR DC for two types of changes. There are design changes to the AP1000 that include the shorter pressurizer, an increase in spent fuel storage within the existing pit and a revision to the bracing of the shield slab below the discharge stack. There are also changes to the FEM to better reflect the structural configuration. The changes that have been incorporated into the dynamic models, in addition to the redesign of the pressurizer compartment, are:

<u>Design changes</u>

- A design change was made in the spent fuel pool area to permit heavier fuel racks. Masses reflecting the racks and spent fuel were updated. In addition, the water in the fuel pits was modeled as lumped masses instead of solid elements.

- The shield building roof slab bracing was modified from tie rods to cross bracing to improve the seismic response.

Model improvements

- The dish model was modified to incorporate changes in the annulus configuration included in existing AP1000 DCD figures. The annulus tunnel on the west side was deleted and replaced by concrete. In addition, nodes and elements were modified in the lower shield building and upper CIS basemat to be compatible with the revised dish model.

- The core makeup tanks (CMTs) were added as stick models.

- Floors in the CIS model were refined to provide better member force results for use in design.

- Polar Crane Model - Changes made to the model weight (3 percent reduction), updated SCV local stiffness, and inclusion of polar crane truck stiffness.

The applicant stated that these changes were considered minor since the NI building basic configuration was not modified. They reflected structural and model changes that were made during design development.

The staff considered RAI-TR03-001 to be resolved, based on the additional description of changes that the applicant added in Revision 1 and Revision 2 of TR-03. However, the applicant subsequently proposed major design changes to the shield building cylindrical wall, air inlets, and roof in "Design Report for the AP1000 Enhanced Shield Building," March 22, 2010. The staff reviewed the most recent revision of TR-03 (Revision 4, March 2010), and noted that the modeling assumptions used in the dynamic models to simulate the new SC cylindrical wall design are not described. Since this is critical information that is not documented in any of the applicant's formal submittals, the staff requested that the details be added to the next revision of TR-03. This was identified as Open Item OI-TR03-01 in the SER with open items.

In its revised response dated August 26, 2010, the applicant stated that the shield building SC modules are modeled by 3D shell elements using modified stiffness and thickness values to simulate equivalent response in the structure. Equations from AP1000 DCD Section 3.8.3.4.1 were provided in response Figure RAI-TR03-001-01, to describe the procedure for calculating equivalent shell element stiffness and thickness values. In its response, in Figure RAI-TR03-001-02, the applicant provided specific values used in the equations. The staff reviewed the equations used and the numerical results obtained, and concluded that the applicant had properly simulated the stiffness of the SC wall in the ANSYS NI10, ANSYS NI20, ANSYS NI05, and SASSI NI20 models. The applicant also provided a proposed revision to TR-03 to incorporate this information. The staff has confirmed that these changes have been incorporated into TR-03. Therefore, RAI-TR03-01 and the associated open item are resolved.

In TR-03, Section 4.0, the applicant discussed the dynamic modeling of seismic Category I structures constituting the AP1000 NI. The staff reviewed the applicant's modeling assumptions with respect to concrete material characterization. For the NI, the applicant stated that the concrete modulus of elasticity was reduced to 80 percent of its nominal value, in order to reduce stiffness to simulate cracking. The staff's review of this section found insufficient technical basis for the 20 percent reduction of the modulus of elasticity. In RAI-TR03-05, the staff requested that the applicant clarify whether this reduced stiffness was used in the dynamic seismic response analyses for generation of FRS, and in the equivalent static acceleration analyses for design of the structural members. If different stiffness assumptions were used, the staff asked the applicant to provide the technical basis. The staff also requested that the applicant provide the technical basis for using 80 percent, by comparing this to guidance in industry documents such as ASCE 4-98, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary,” and to describe any sensitivity studies conducted to determine the effect of varying the concrete stiffness on ISRS and design of structural members.

In its response dated January 18, 2007, the applicant stated that the reduction to 80 percent is to account for the effects of cracking, as recommended in Table 6-5 of FEMA 356 (Reference: FEMA 356, “Pre-standard and Commentary for the Seismic Rehabilitation of Buildings,” (FEMA, November 2000) and that the staff had accepted this basis as part of the AP1000 DCD, Revision 15 review.

The staff evaluated the response and confirmed that during the DC review of the AP1000 founded on HR, the staff had accepted FEMA’s recommendation regarding the application of a structural stiffness factor of 0.8 for the seismic analysis of the NI structures.

During the April 2007 audit, the staff requested that the applicant revise its response to clarify that the 0.8 factor for concrete stiffness correlates with test results for essentially uncracked concrete, and does not account for observed or predicted significant cracking (for which a 0.5 factor is more appropriate).

In its revised response dated June 15, 2007, the applicant added that the reduction to 80 percent reflects the observed behavior of concrete when stresses do not result in significant cracking. The applicant also proposed a revision to TR-03, Section 4.0, indicating that concrete structures are modeled with linear elastic uncracked properties, but the modulus of elasticity is reduced to 80 percent of its value to reduce stiffness, to reflect the observed behavior of concrete when stresses do not result in significant cracking, as recommended in Table 6-5 of FEMA 356.

The staff evaluated the response and accepted the applicant’s clarification that the use of 0.8 stiffness factor applies when stresses do not result in significant cracking. The staff confirmed that the changes were properly documented in TR-03, Revision 1.

Subsequent to the resolution of RAI-TR03-05, the applicant made major design changes to the cylindrical wall, air inlets, and roof of the shield building. The staff’s separate review of the shield building redesign raised questions about the acceptability of the 0.8 factor, since preliminary results presented by the applicant indicate that significant concrete cracking occurs in some areas under seismic loading. The staff requested that the applicant study the sensitivity of the shield building seismic response to a 0.5 stiffness reduction, which is more appropriate when there is significant concrete cracking. The staff had concern that significant concrete cracking could shift the fixed-based frequencies of the shield building, potentially leading to an increase in the seismic demand on the shield building structure and on any systems and components attached to the shield building structure. In its review of TR-03, Revision 4 (March 2010), the staff noted that the 0.8 factor was used for the shield building reanalysis without any discussion or technical justification. This issue was identified as Open Item OI-TR03-05.

In its response dated August 3, 2010, the applicant stated that OI-TR03-05 is addressed in the response to OI-SRP3.7.1-SEB1-19. The staff reviewed the applicant’s response to OI-SRP3.7.1-SEB1-19, dated August 26, 2010, and confirmed that it addresses the use of a

0.8 factor for concrete modulus in the design-basis linear seismic analyses. To demonstrate that using a reduced concrete modulus of 0.8 x Ec in the design-basis seismic analysis of the NI is appropriate to account for stiffness reduction due to concrete cracking, the applicant performed nonlinear ABAQUS analysis, using a smeared concrete cracking model, and compared the results to the results of a linear ABAQUS analysis, which assumed 0.8 x Ec for the concrete modulus. The applicant submitted additional details of this comparison in its response to related OI-SRP3.8.3-SEB1-03.

The applicant compared the ABAQUS results (linear and nonlinear) to linear ANSYS NI20 results, in order to validate that the ABAQUS models are dynamically similar to the ANSYS design-basis model. The applicant presented response spectra comparisons, in three orthogonal directions, at the shield building roof in Figures RAI-SRP3.7.1-SEB1-19-11 through RAI-SRP3.7.1-SEB1-19-13 of the response. The comparisons show that the nonlinear ABAQUS model results are very similar to and enveloped by the linear model results, which assume 0.8 x Ec. The applicant also provided a plot of stress-strain for a highly stressed element in the shield building (West wall location), in Figure RAI-SRP3.7.1-SEB1-19-02 of the response. The applicant stated that while principal stress values are at or near the assumed cracking threshold 2.1 MPa (43 ksf), the concrete strains are relatively small; and further stated that the associated secant stiffness would be close to 0.8 x Ec, as shown in Figure RAI-SRP3.7.1-19-01 of the response.

The staff reviewed the applicant's analysis results presented in the response to this RAI and in the response to OI-SRP3.8.3-SEB1-03, and determined that the applicant has provided a sufficient technical basis for using a reduced concrete modulus of 0.8 x Ec to account for stiffness reduction due to cracking. The response is acceptable on the basis that the applicant's comparison of linear (0.8 x Ec) and nonlinear (concrete cracking model) analysis results showed a very good correlation, with the linear model being conservative. Therefore, RAI-SRP3.7.1-SEB1-19 is resolved. On the basis that OI-SRP3.7.1-SEB1-19 is resolved, OI-TR03-005 is also resolved.

In TR-03, Section 4.1, the applicant described the modeling assumptions used in the seismic analysis for the water inside the passive containment cooling water storage tank (PCCWST) on the shield building roof. The applicant indicated that a significant percentage of the water mass responds at very low frequency (sloshing), and does not affect the overall building seismic response. Consequently, the applicant concluded that the sloshing water mass could be excluded in the two horizontal directions.

The staff's review of this section found that there was insufficient basis for accepting the applicant's exclusion of sloshing water mass in the dynamic analysis models. In RAI-TR03-007, the staff requested that the applicant provide a detailed technical basis for excluding the low-frequency, water-sloshing mass and to quantify the percentage of water mass in the PCCWST that was excluded.

In a letter dated January 29, 2007, the applicant stated that sloshing of the water in the AP1000 PCCWST was analyzed using a formula for toroidal tanks (Reference J.S. Meserole, A. Fortini, "Slosh Dynamics in a Toroidal Tank," Journal Spacecraft, Volume 24, Number 6, November-December 1987). The fundamental sloshing frequency given by the formula is 0.136 Hz with a modal mass equal to 65 percent of the water mass.

The applicant further stated that AP600 analyses by formula gave frequencies and effective masses similar to those in the AP1000 analyses, and the sloshing formula was confirmed for the

AP600 by analyses of a 3D FEM of the water in a rigid tank. For the AP600 design models of the ASB, the applicant found that:

- 60 percent of the water mass was in a sloshing mode. This was included in the AP600 stick model at the elevation of the tank with two masses each with 2 horizontal degrees of freedom.
- The total sloshing mass is 2.6 percent of the mass of the ASB. The stick model results show a maximum absolute acceleration of the sloshing masses of 0.13g, at a frequency of 0.136 Hz.
- The fundamental frequency of the ASB is between 2 and 3 Hz, and the acceleration is 1.1g at the base of the tank.

As a result of the above, the applicant concluded that the low-frequency sloshing mode is not significant to the response of the NI away from the shield building roof and that this conclusion could be extended to the AP1000 design. The horizontal mass participating in the sloshing mode was excluded from the AP1000 3D shell dynamic model of the shield building. However, the applicant considered sloshing in the hydrodynamic loads for the tank wall design.

The staff reviewed the applicant's response and discussed it with the applicant during the April 2007 audit. The applicant stated that the effect of the low-frequency sloshing mode was confirmed to be negligible by performing an analysis of the AP1000 NI stick model without the low-frequency mass, and comparing these results to the results obtained with the low-frequency masses included, provided in Revision 15 of the AP1000 DCD. Comparisons of maximum absolute accelerations, member forces, and FRS indicated there were no significant changes in any of the responses. The staff reviewed the tank sloshing reference and the applicant's calculation. The staff questioned why the percentage of sloshing mass does not go down for the AP1000 versus the AP600, since the increased volume is achieved primarily by making the tank deeper. The applicant agreed to check its estimate of sloshing mass, and provide its conclusions in a supplemental response.

In its revised response dated July 5, 2007, the applicant provided the key dimensions, frequencies and effective masses of the AP600 and AP1000 tanks as shown below.

Parameter	AP600	AP1000	Units
Inside radius of tank	5.3 (17.5)	5.3 (17.5)	m (ft)
Outside radius of tank	11.6 (38.0)	12.9 (42.5)	m (ft)
Average water depth	6.355 (20.85)	6.92 (22.7)	m (ft)
Sloshing frequency	0.139	0.136	Hertz
Ratio of sloshing to total mass	0.66	0.65	none

The staff evaluated the response, and concluded that the explanation provided by the applicant to address why the sloshing mass ratio remained unchanged between AP600 and AP1000 was acceptable.

The applicant subsequently made design changes to the PCCWST on top of the shield building. The staff noted that the applicant needed to recalculate the sloshing frequency and sloshing mass to account for any changes in the tank geometry, water depth, and/or free board above the water surface. The staff had concern that overestimating the water sloshing mass could result in an under-prediction of seismic demand for the tank structure. This issue was identified as Open Item OI-TR03-07.

In a letter dated July 12, 2010, the applicant submitted a supplement to its previous RAI-TR03-07 response, stating that the dimensions of the PCCWST were not changed in the enhanced shield building design. The only change affecting the PCCWST is a reduction in elevation by about 1.52 m (5 ft). The applicant also conducted an updated fluid sloshing analysis of the PCCWST, using an ANSYS model of the fluid in a rigid tank. The results of the ANSYS analysis support the 60 percent assumption for low frequency sloshing modes, as shown below.

Parameter	AP1000		Units
Water weight in 180 degree model	1.154×10^6 (3,337)		kg (kip)
Frequency	0.119	0.321	Hertz
Participating weight	$7{,}253 \times 10^5$ (1,599)	1.623×10^5 (358)	kg (kip)
Ratio of sloshing to total mass	47.93	10.73	%

The staff evaluated the applicant's updated analysis results, and concluded that the PCCWST response has a very significant water sloshing component, which has a negligible effect on the overall seismic response of the ASB. On this basis, OI-TR03-07 is resolved.

3.7.2.4 Soil-Structure Interaction

The staff performed a review of the applicant's SSI analyses described in AP1000 DCD Section 3.7.2.4, AP1000 DCD Appendix 3G, and TR-03, using the guidance in NUREG-0800 Section 3.7.2. The design-basis SSI analyses use the AP1000 CSDRS as the seismic input motion; the acceptability of these analyses is evaluated in Section 3.7.2.4.1 of this report. The staff also performed a review of the applicant's evaluation of HRHF ground motion effects described in AP1000 DCD Appendix 3I and TR-115. Since the staff addressed special considerations for seismic evaluation of HRHF sites in NUREG-0800 Section 3.7.2, under acceptance criteria for SSI, the staff has included the HRHF evaluation in Section 3.7.2.4.2 of this SER.

3.7.2.4.1 Nuclear Island Seismic Analyses using CSDRS Input Motion

In AP1000 DCD Section 3.7.2.4, the applicant stated that the SSI analyses for the FR and soil sites are described in AP1000 DCD Appendix 3G. In AP1000 DCD Sections 3G.4.1 and 3G.4.2, the applicant described the 3D SSI and fixed based analyses. Additional details of these analyses are described in TR-03.

The applicant performed SSI analyses using the computer program SASSI and the NI20 3D finite element shell model. The SSI analyses were performed for the five soil conditions described in AP1000 DCD Section 3G.3, and reviewed in Section 3.7.1.3 of this SER. The SASSI model included a surrounding layer of excavated soil, as shown in AP1000 DCD Figures 3G.4-3 and 3G.4-4. The seismic input consisted of three statistically independent acceleration time histories (north-south, east-west, and vertical directions), each applied separately. The three resulting time history responses (one for each direction) are combined algebraically at each instant in time. AP1000 DCD Figures 3G.4-5X through 3G.4-10Z provide comparisons of ISRS for the soil cases analyzed. The applicant also performed fixed-base analysis using the ANSYS NI20 model, to simulate HR conditions (i.e., Vs greater than 2438 m/s (8,000 fps)).

Selection of Soil Cases

The staff reviewed the applicant's description of site studies and selection of soil cases described in Section 4.4.1.2 of TR-03. The staff's review of Tables 4.4.1-1A and 4.4.1-1B of TR-03 identified that the applicant used three soil/rock degradation models in its parametric studies for selecting site conditions: Seed and Idriss 1970 soil/rock degradation curves; Idriss 1990 soil degradation curves; and EPRI 1993 soil degradation curves. In RAI-TR03-10, the staff requested that the applicant provide the technical basis for using these different soil degradation models for its parametric studies.

In its response dated January 18, 2007, the applicant stated that SSI analyses on rock sites for both the AP600 and the AP1000 use the rock degradation curve recommended by Seed and Idriss (Reference: Seed, H.B. and I.M. Idriss, "Soil Moduli and Damping Factors for Dynamic Response Analysis," Report Number. EERC [Energy and Environmental Research Center] 70-14, Earthquake Engineering Center, University of California, Berkeley, CA, 1970). This was applied in SSI analyses for the HR, FR and SR sites. The applicant further stated that SSI analyses on soil sites for the AP1000 used the latest soil degradation curve recommended by EPRI (Reference EPRI TR-102293, "Guidelines for Determining Design Basis Ground Motions," 1993). This was applied in SSI analyses for the UBSM, SM, and SS sites. Two sets of degradation curves were used in the AP600 studies. The early analyses used the degradation curve recommended by Seed and Idriss. Later analyses performed to address NRC questions used the later soil degradation curve recommended by Idriss (Reference Idriss, I.M., "Response of Soft Soil Sites during Earthquakes," H. Bolton Seed Memorial Symposium Proceedings, May 1990). The applicant provided a proposed revision to AP1000 DCD Section 3.7.1.4 and additional figures for inclusion in the AP1000 DCD.

The staff evaluated the response and noted a number of issues in need of further clarification:

1. The EPRI 1993 model shown in the proposed AP1000 DCD Figure 3.7.1-16 indicates hysteretic damping levels greater than 15 percent. In NUREG-0800 Section 3.7.2.4, the staff imposed a limit of 15 percent on hysteretic damping. The applicant should provide the final iterated Vs profile and damping levels reached throughout the soil column, for each case analyzed for site response, and show that damping levels do not exceed the 15 percent limit.

2. The EPRI 1993 model is generally considered appropriate for cohesionless soils. The model is not considered appropriate for cohesive fine-grained soils. The AP1000 DCD should indicate the criteria to be used by the COL applicant to evaluate the appropriateness of this degradation model for site-specific application.

3. The AP1000 DCD should include the strain-iterated Vs profiles that need to be compared to the site-specific velocity profiles generated by the COL applicant.

During the April 2007 audit, the applicant agreed to supplement its response by identifying the bounds of the strain-iterated Vs profiles. The applicant also agreed to describe how a COL applicant confirms that its site is enveloped by the generic seismic design basis. In its revised response dated July 5, 2007, the applicant stated that: (1) the soil profiles used in the generic analyses will be added to AP1000 DCD Section 3.7.1.4; (2) additional clarification of how to confirm that a specific site is enveloped by the generic seismic design basis will be provided in proposed revisions to AP1000 DCD Section 2.5.2; and (3) TR-03, Section 4.4.1.2, will be revised to include the description and table of degraded properties for each soil profile.

During the May 2008 audit, the staff and the applicant agreed that the site-specific Vs profile should be based on low-strain minimum measured values; and that a criterion is needed to define the acceptable variation in Vs when the site-specific soil profile shows an inversion (i.e., soft material under hard material). These issues are addressed under RAI-SRP2.5-RGS1-15.

During the April 2009 audit, the staff requested that the applicant provide clarification in the AP1000 DCD concerning limitations on the use of two dimensional (2D) SASSI analyses to address site-specific deviations from the certified design site parameter envelope. In a letter dated May 15, 2009, the applicant submitted a proposed revision to AP1000 DCD Section 2.5.2.3 to provide this clarification:

> The Combined License applicant may identify site-specific features and parameters that are not clearly within the guidance provided in subsection 2.5.2.1. These features and parameters may be demonstrated to be acceptable by performing site-specific seismic analyses. If the site-specific spectra at foundation level at a hard rock site or at grade for other sites exceed the certified seismic design response spectra in Figures 3.7.1-1 and 3.7.1-2 at any frequency (or Figures 3I.1-1 and 3I.1-2 for a hard rock site), or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed. These analyses may be either 2-D or 3-D.
>
> - 3-D SASSI analyses will be used to quantify the effects of exceedances of site-specific GMRS compared to the CSDRS, or the HRHF GMRS at a hard rock site (DCD Figures 3I.1-1 and 3I.1-2), or in cases where the site specific velocity soil profiles do not fall within the range evaluated for the standard design.
>
> - 2-D analyses are performed for parameter studies.
>
> - Results will be compared to the corresponding 2-D or 3-D generic analyses.

The staff reviewed the applicant's proposed revision to AP1000 DCD Section 2.5.2.3, and the applicant's response to RAI-SRP2.5-RGS1-15, and concluded that the open technical issues had been adequately addressed. The applicant clarified the limitations on the use of 2D [ ] analyses to address site-specific deviations from the certified design site parameter envelope; and also provided additional criteria that must be satisfied at a specific site in order to be covered by the AP1000 generic soil site analyses. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In Section 4.4.1 of TR-03, the applicant stated that many results and conclusions from the AP600 soil studies are applicable for the AP1000. In RAI-TR03-14, the staff requested that the applicant describe which results and conclusions from the AP600 soil studies are applicable to the AP1000.

In a letter dated January 18, 2007, the applicant stated that the AP600 design is based on enveloped results from analyses for four soil conditions (HR, SR, UBSM, and SM). These four soil cases were selected from the parametric analyses summarized in Section 4.4.1 of TR-03. The AP600 soil studies demonstrated that these four cases would bound sites having soil with Vs exceeding 1,000 fps. Parameters selected for the design soil cases from these analyses were:

- Depth to bedrock of 36.7 m (120 ft)
- Water table for the UBSM and SM cases up to grade
- Parabolic variation of Vs with depth for the UBSM and SM cases

The applicant stated that parametric analyses of the AP1000 were performed for six soil cases, as described in TR-03, Section 4.4.1.2. These analyses used the same assumptions for depth-to-bedrock, depth-to-water table, and variation of Vs with depth as used in the AP600 analyses. These analyses confirmed that the response of the AP1000 was similar to that of the AP600 for these soil cases, with the AP1000 fundamental response occurring at lower frequencies due to its increased height 7.6 m (25 ft) and mass (10 percent).

The staff evaluated the RAI response and concluded that the applicant provided a sufficient description of the design parameters derived from the AP600 analyses in TR-03, Section 4.4. On this basis, RAI-TR03-14 was resolved.

In TR-03, Section 4.4.1, the applicant concluded that some effects (water table, soil layering, soil-degradation model, etc.) are not significant for the seismic response of the NI structures. The staff's review of this section found that the applicant did not provide sufficient basis for making the above conclusions. In RAI-TR03-15, the staff requested that the applicant provide the technical basis for drawing these conclusions for the AP1000. In addition, the staff requested that the applicant demonstrate that the combination of these effects is also insignificant for the seismic response of the NI structures.

In a letter dated August 20, 2008, the applicant submitted a comprehensive response to address the staff's questions. The referenced figures and tables were submitted as part of the RAI response. Paraphrasing the applicant's response:

> Revised TR-03 Section 4.4.1.1 provides additional technical basis for the selection of the soil parameters used in the AP1000 3D SASSI design cases. The soil cases selected for the AP1000 use the same parameters on depth-to-bedrock, depth-to-water table and variation of Vs with depth as those used in the AP600 design analyses. The parameters used for the AP1000, based on the results and conclusions from the AP600 soil studies, are summarized in Table 4.4.1-1A. The AP600 soil studies considered variations of the parameters and combinations thereof in establishing the design soil profiles. AP1000 has a footprint identical to that of the AP600 and is similar in overall mass. The height of the shield building is increased by about 6.1 m (20 ft). The total weight of the NI increases by about 10 percent. Parametric analyses of the AP1000 were performed for six soil cases, as described in Section 4.4.1.2. The AP1000 response is very similar to AP600, except that the fundamental response occurs at lower frequencies due to the

increased height and mass of the NI. Based on the similar response in these analyses, it is concluded that the governing parameters obtained for the AP600 soil studies are also applicable to the AP1000.

The applicant addressed soil degradation in RAI-TR03-10. Tables of strain-iterated Vs used in the generic analyses are shown in Table 4.4.1-3 of TR-03.
Figure RAI-TR03-15-1 shows the bounds of these strain-iterated Vs profiles. The combination of effects of the different soil parameters is reflected in these bounds. Figure RAI-TR03-15-2 shows how a COL applicant could demonstrate that the site is enveloped by generic seismic design basis. The applicant would define its site geotechnical parameters as defined in AP1000 DCD Section 2.5 and would justify why the site is within the bounds of the AP1000 generic analyses that have been considered in this TR. These parameters would include the soil profiles used in the probabilistic seismic hazard analysis (PSHA), which could then be compared to Figure RAI-TR03-15-1. Subsequent discussions between the COL applicant and the NRC may uncover a parameter for which more justification is required, in order to show that the impact of this parameter on the response is small. This justification could be done with the AP1000 2D model. An example of how a 2D parametric study would be used is shown in Figure RAI-TR03-15-3 and RAI-TR03-15-4. If the parametric 2D SASSI studies show that the effect could be significant (e.g., 90 percent of the design spectrum, see Figure RAI-TR03-15-4) when compared to the 2D design spectra, a 3D SASSI study would then be performed. If the 3D SASSI analyses show some exceedances at the critical locations, the applicant would then proceed to show that sufficient margin exists in the design to accommodate these exceedances.

The effect of water table on the seismic response of the NI structures is shown in Figures RAI-TR03-15-5 through RAI-TR03-15-7. Case 1 (SM) shows the results for the SM generic case profile, which assumes water table at grade. Case 2 (SM-NW) results are for the same soil condition except the water table is below the bottom of the soil profile at 36.7 m (120 ft) below grade. As can be seen, there is negligible difference between the two cases for the horizontal response. The vertical response due to the design profile with the water table at grade (Case 1) is more conservative than that for the dry soil profile (Case 2). This result is similar to the results in the AP600 study, which are summarized in TR-03, Section 4.4.1.1. Thus, the generic analyses are conservative for sites with a lower water table.

The staff determined that the information presented in the applicant’s revised response to RAI-TR03-15, and supplementary information in the RAI-TR03-10 response related to soil degradation models, are sufficient to address the staff’s questions. The staff also confirmed that all proposed revisions to TR-03 have been formally submitted in Revision 4. Therefore, RAI-TR03-15 is resolved.

Seismic Analysis Results

During its review of TR-03, the staff identified that equivalent static analysis was employed to calculate maximum member forces for detailed design of the NI structures, using acceleration versus height profiles obtained from the time history analyses. The staff’s separate review of TR-09, “Containment Vessel Design Adjacent to Large Penetrations,” identified that the SCV is designed for equivalent static accelerations determined from the fixed-base NI stick model, tabulated in AP1000 DCD Table 3.7.2-6, which are representative of the HR condition. In RAI-TR03-16, the staff requested that the applicant: (1) identify the site condition(s) selected to develop the equivalent static acceleration profile used to perform the equivalent static analysis; and (2) discuss whether the seismic loads used for design of the SCV envelop both the fixed-base HR condition and the worst-case condition from all soil sites considered.

The applicant's initial responses to this RAI did not fully address the staff's concern. As an alternative, the staff requested that the applicant provide a direct comparison of the equivalent static analysis results to time history analysis or RSA results. During the October 2007 audit, the applicant indicated it had switched the detailed evaluations of the CIS and ASB from equivalent static analysis to RSA. However, for the SCV, the applicant did not address whether the equivalent static acceleration method yields conservative results, when compared to RSA or time history analysis.

At the April 2009 audit, the applicant presented a comparison of results for the SCV, between equivalent static analysis and a mode superposition time history analysis, at major containment penetrations. The comparison showed that the equivalent static analysis results are higher than the time history results. The applicant agreed to revise its RAI response, to include the information presented at the audit.

In its revised response dated May 15, 2009, the applicant stated that the equivalent static acceleration analyses of the containment vessel (CV), described in TR-09, use a finite element shell model with a refined mesh in the area adjacent to the large penetrations (Figure 2-6 of TR-09). A reanalysis was performed using the same methodology on the coarse-mesh model of the SCV. The applicant performed a time history analysis of the coarse-mesh model, selecting information for the regions immediately surrounding the large penetrations, as shown in Figure RAI-TR03-016-001, for the purpose of comparing the loads from equivalent static analysis and time history analysis. The effects of the missing mass in the time history analysis were incorporated by an algebraic sum of the stress intensities from a run with the left-out mass accelerated at zero period acceleration (ZPA) and the modal superposition time history analysis. Figures RAI-TR03-016-002 through RAI-TR03-016-005 (attached to the RAI response) compare the stress intensity for individual elements surrounding the major penetrations. The applicant stated that the results from these analyses show that equivalent static analysis consistently produced higher stresses than the time history results. The staff reviewed the analysis comparisons and concluded that the equivalent static acceleration results for the SCV are conservative, when compared to time history results. Therefore, RAI-TR03-16 is resolved.

During its review of Section 6.2 of TR-03, the staff identified a number of editorial and technical items in need of clarification or explanation. In RAI-TR03-21, parts (b), (c), and (e), the staff requested that the applicant provide technical clarifications. Parts (a) and (d) were editorial.

> (b) TR-03, Section 6.2, states "For those local flexible structures that are amplified, apply an additional acceleration to these structures equal to the difference between the average uniform amplified component accelerations and rigid body component equivalent static accelerations. These accelerations are to be considered in local design of the flexible portion of the structure but do not need to be considered in areas of the structure away from the local flexibility. They can be applied in a series of individual load vectors." The applicant has not shown how this methodology has been implemented, and whether the effects of increased accelerations on locally flexible structures can be ignored in areas of the structure away from the locally flexible structures. The sum total of all the flexible masses times the corresponding acceleration increments may impose greater-than-negligible additional loads on the overall structure, in the two horizontal directions and in the vertical direction. Therefore, the applicant is requested to

(1) describe in greater detail the implementation of this methodology, including a numerical example; and (2) provide a quantitative technical basis for the conclusion that the effects of increased accelerations on locally flexible structures can be ignored in areas of the structure away from the locally flexible structures."

(c) TR-03, Section 6.2, states "The vertical equivalent static seismic accelerations at (Shield Bldg) elevations 89.9 m (294.93 ft) and 101.5 m (333.13 ft) are obtained directly from the maximum time history results by taking the average of locations at opposite ends of a diameter. The vertical accelerations from the 3D finite element model at the shield building edges at these elevations are significantly influenced by the horizontal loading. If they are used for the vertical equivalent accelerations, the horizontal response would be double counted in the vertical direction." The applicant has not shown how this methodology has been implemented or its basis. Therefore, the applicant is requested to submit a numerical example, based on elevation 101.5 m (333.13 ft) of the SB, to demonstrate the implementation of this methodology. In this example, please also include the vertical acceleration value that would be obtained if this methodology was not implemented."

(e) TR-03, Section 6.2, under the heading "Seismic Accelerations for Evaluation of Building Overturning," states "The dynamic response of the structure affecting overturning and basemat lift off is primarily the first mode response at about 3 Hertz on hard rock. This reduces to about 2.4 Hertz on soil sites as shown in the 2D ANSYS and SASSI analyses. The higher auxiliary building accelerations of Table 6.2-2 are not considered in overturning since they are from higher frequency modes greater than 2.4 Hertz. Amplified response of individual walls in the Auxiliary Building and the IRWST [In-Containment Refueling Water Storage Tank] need not be considered since they are local responses that do not effect overturning." For the overturning analysis, the staff is concerned that the methodology employed may not predict an overall moment on the basemat that envelops the maximum overturning moment for all site conditions. The applicant is requested to provide its technical basis for the conservatism of the methodology employed.

In a letter dated April 5, 2007, the applicant provided its initial response to this RAI. For part (b), the staff required additional clarification concerning how the applicant determined the uniform acceleration values applied to the whole structure and the additional acceleration increments applied to the flexible areas.

For part (c), the applicant stated that a seismic component associated with the rotational response of the PCCWST should also be included, in addition to the translational seismic acceleration component, and that the rotational response of the PCCWST would be addressed in the redesign of the shield building roof.

For part (e), the applicant proposed that it be deferred to the staff's review of TR-85, APP-GW-GLR-044, Revision 0, "Nuclear Island Basemat and Foundation."

At the October 2007 audit, partly in response to part (b) of this RAI, the applicant presented results from an RSA of the coupled ASB/CIS, using the refined ANSYS NI05 model. The applicant had decided to use these RSA results as the basis for detailed design of the ASB and CIS. At the time, the applicant stated that switching to RSA resolved parts (b) and (c) of this RAI.

During the May 2008 audit, the staff requested that the applicant demonstrate that the seismic RSA using the fixed base NI05 model is sufficient to capture additional amplification due to rocking. The applicant agreed to compare loads at the top of the shield building, between time history analysis, which includes rocking, and RSA, which does not.

On August 20, 2008, the applicant submitted its revised response to parts (b) and (c) of this RAI. The staff concluded that the questions raised in part (b) of this RAI were no longer applicable. The staff confirmed that TR-03, Revision 2, Section 6.4, clearly identified that RSA is used for the ASB design and the CIS design. Therefore, part (b) was resolved.

For part (c), the applicant presented a comparison of the bending moments in the beams at the top of the shield building, and the forces and moments in the PCS vertical wall, between time history and RSA results. In all cases, the RSA is conservative when compared to the time history analysis, confirming that conservatism in the RSA that will account for rocking. The staff concluded that the comparisons sufficiently demonstrated the conservatism of the RSA results. Therefore, part (c) was resolved.

Part (e) of this RAI, concerning the conservatism of the overall moment on the basemat, is addressed in Section 2.6.1.2 of TR-85 and is tracked under the staff's TR-85 evaluation. This issue is considered resolved with respect to the TR-03 evaluation. Therefore, RAI-TR03-21 was resolved.

The staff reviewed the applicant's seismic displacement results presented in TR-03, Section 6.3. The maximum seismic deflections obtained from the fixed-base time history analysis and the SASSI analyses are given in Tables 6.3-1 to 6.3-3 for the ASB, CIS, and SCV, respectively. The staff determined that a number of clarifications were needed before the staff could complete its review. In RAI-TR03-22, the staff requested that the applicant: (1) clarify whether the deflections in the tables are a consistent set, based on the worst-case time history result, or are an envelope of maximum deflections from all the time history results; and (2) compare the tabulated deflections to the corresponding deflections obtained from the equivalent static acceleration analyses, and explain any significant differences.

In its response dated January 29, 2007, the applicant stated that the deflections given in Tables 6.3-1 to 6.3-3 are the envelope of maximum relative deflections from all of the time history results for the soil and HR cases. Displacements at different nodes for the soil cases have been obtained relative to the translation of a reference node at the bottom of the foundation and near the center of the basemat. Deflections for the HR case are relative to the fixed base at foundation level.

The applicant further stated that the deflections given in these tables have been revised to remove drift, by adding a small constant acceleration to the response acceleration at every time step for the first 0.05 seconds of the time history. If baseline correction is not performed, a residual drift in displacement time histories will be obtained at the end of the seismic excitation. The applicant provided Tables RAI-TR03-022-1 to RAI-TR03-022-3 in its response, showing the revised relative displacements. The applicant also stated that it is not possible to compare equivalent static displacements to the time history displacements for the soil cases. The time history results include rocking about the base, while the equivalent static analysis has a fixed base.

The staff questioned the approach the applicant had used to eliminate drift and, following discussions of this issue during audits in 2007, and 2008, the applicant submitted a revised

RAI response, in a letter dated August 20, 2008. The applicant revised the approach for eliminating drift. The new approach calculates displacements internally within the SASSI program, based on an analytical complex frequency domain approach that uses inverse fast Fourier transforms to compute relative displacement histories, instead of double numerical integration in the time domain for computing absolute displacement time histories from absolute acceleration time histories. The analytical approach is more accurate than a typical baseline correction (time integration) algorithm. The applicant also submitted a proposed revision to TR-03, Section 6.3, "Seismic Displacement Calculation," adding more detail about the analysis methodology and identifying that the ACS SASSI RELDISP module is used for this calculation.

The applicant also indicated in its response that it had switched to seismic RSA and is not using equivalent static analyses; and consequently the staff's initial request for comparison of dynamic results to equivalent static analysis results is no longer applicable. The applicant also submitted a proposed revision to TR-03, Section 6.3, covering this change.

The staff reviewed the response and found the applicant's revised approach to eliminate drift acceptable, because it is mathematically rigorous. For comparison of displacements, the staff noted that RSA is only applied to the ASB and CIS, not to the SCV. Thus, this issue remained unresolved for the SCV. The staff confirmed that TR-03 had been appropriately revised in Revision 3, resolving the drift issue. The applicant also submitted a detailed comparison of time history results to equivalent static acceleration results for the SCV, in a revised response to RAI-TR03-16, demonstrating the conservatism of the equivalent static analysis for the SCV. As a result, the staff considered the static versus. dynamic issue resolved for the SCV. Therefore, RAI-TR03-22 was resolved.

In a letter dated September 10, 2010, the applicant submitted revised responses to RAI-TR03-22 and related RAI-TR03-37. These responses identified alternate methods that the applicant has used to calculate relative displacements. The applicant identified two methods, in addition to the ACS SASSI RELDISP module, for inclusion in the next revision of the AP1000 DCD and the next revision of TR-03. The proposed AP1000 DCD additions, included in the response to RAI-TR03-37, are as follows:

DCD 3G.4.1 "ANSYS Fixed Base Analysis"

> ANSYS is used to calculate the maximum relative deflection to the nuclear island for the envelop case that considers all of the soil and hard rock site cases. Synthesized displacement time histories are developed using the envelope seismic response spectra from the six site conditions (hard rock, firm rock, soft rock, upper-bound soft-to-medium, soft-to-medium, and soft soil). Seismic response spectra at nine locations are used (4 edge locations, 1 center location, and 4 corner locations). It is not necessary to adjust for drift since deflections relative to the basemat are calculated, and the drift would be subtracted from the results.

DCD 3G.4.2 "3D SASSI Analyses"

> Westinghouse has adopted the approach that calculates displacements internally within the ACS SASSI program based on an analytical complex frequency domain approach that uses inverse Fast-Fourier Transforms (FFT) to compute relative displacement histories instead of double numerical integration in the time

domain that computes absolute displacement time histories from absolute acceleration time histories.

The relative displacement time history is calculated using ACS SASSI RELDISP module. The complex acceleration transfer functions (TF) are computed for reference and all selected output nodes. The relative acceleration transfer function is calculated by subtracting the reference node TF from the output node TF. The relative displacement transfer function is obtained by dividing the circular frequency square (ω^2) for each frequency data point. The relative displacement time history is obtained by taking the inverse FFT.

Relative displacements are calculated between adjacent buildings and the nuclear island using soft springs between the buildings. The spring stiffness is very small so that it does not affect the dynamic response. These calculations are performed using 2-D models and the SASSI 2000 code. The relative deflection is calculated using the maximum compressive spring force and the stiffness value.

The applicant also proposed comparable revisions to TR-03 in the response to RAI-TR03-22. The staff determined that the additional methods used by the applicant to calculate relative displacements are technically correct, and do not require any correction for drift. In subsequent revisions to the AP1000 DCD and TR-03, the applicant made appropriate changes which resolve this issue.

The staff reviewed the comparison of the NI10 and NI20 seismic analysis models, described in TR-03, Appendix C. The staff's review identified the need for a number of clarifications and explanations of the results presented. In RAI-TR03-32, the staff requested that the applicant provide these clarifications and explanations.

The staff and the applicant discussed the issues raised in this RAI at audits in 2007, 2008, and 2009. The applicant submitted several revisions to its RAI response, to address the staff's original and follow-up questions. Following the April 2009 audit, the only remaining technical issue was whether the NI20 model refinement is sufficient to represent vibration modes up to 33 Hz are potentially excited by the CSDRS ground spectrum input. The staff was concerned that, if the dynamic analysis model(s) of the AP1000 do not accurately predict the amplified response of flexible regions, then the ISRS at those locations may be underestimated. The staff initiated an independent comparison of modal properties between the ANSYS NI10 model and the ANSYS NI20 model. Based on the preliminary results of the staff's confirmatory analyses, the staff requested that the applicant demonstrate that all walls, floors, and roof slabs with a fundamental plate vibration frequency less than 33 Hz are adequately represented in the NI20 model, such that an ANSYS NI20 modal analysis will capture these vibration modes. If this is not the case for specific walls, floors, or roof slabs, the staff requested that the applicant develops an approach to generate the ISRS that consider the additional amplification in the middle of the wall, floor, or roof slab.

In TR-03, Revision 4 (March 2010), Section 4.2.4, the applicant stated that the NI05 model was reviewed to identify flexible regions that may produce amplified response spectra. The applicant concluded that the NI20 model was too coarse in some areas to pick up all local vibration modes up to 33 Hz, based on comparison to NI05 modal analysis results.

Consequently, the seismic response in the middle of some wall, floor, and roof panels is underestimated, leading to nonconservative ISRS for subsystem design. To address this, the applicant proposed a method of evaluating these areas using the more detailed NI05 model to evaluate flexible regions. The staff's review of the proposed method found that that there was insufficient description of the proposed method and that an example case (including results) would be helpful in understanding the implementation. This issue was identified as Open Item OI-TR03-32.

In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-TR03-32. The applicant stated the NI05 model had been reviewed for flexible regions where out-of-plane response may occur at frequencies less than 33 Hz. The applicant noted that each of the regions reviewed have a higher mesh refinement than the NI20 model. The regions, which have flexible areas, are evaluated in one of two ways:

1. Flexible areas that were previously identified (TR-03, Revision 4, Table 4.2.4-10) have amplified response spectra developed from the envelope of the time history analysis results for the HR and soil sites.

2. Flexible regions, which require a detailed analysis to obtain the amplified response spectra use input directly from time history analysis. The NI05 finite element model is used to capture out-of-plane flexibilities that, because of mesh refinement, the NI10 and NI20 models could not capture. The resulting nodes have been designated with (NI05) to distinguish that the amplified response spectra come from that model.

This applicant identified proposed revisions to TR-03, to document the methods and results. The staff reviewed the flexible regions identified in Tables RAI-TR03-032-2, RAI-TR03-032-3, and RAI-TR03-032-4 of the RAI response, and the ISRS comparisons (NI05 amplified versus NI10/NI20) shown in Figures RAI-TR03-032-7 to RAI-TR03-032-13 of the RAI response. Based on its review, the staff finds the applicant's method for identifying flexible regions and modifying the ISRS to be acceptable. By using the mesh refinement of the NI05 model, the applicant was able to locate and evaluate flexible regions of the NI structures that were inadequately modeled in the less refined NI20 and NI10 models. RAI-TR03-32 and the associated open item are resolved. In a subsequent revision to TR-03, the applicant made appropriate changes to the report text. RAI-TR03-32 and the associated open item are resolved.

The staff reviewed TR-03, Section 4.2.4, which summarizes the applicant's seismic analysis models and methods used for the AP1000 design. In Table 4.2.4-1 of TR-03, the applicant summarized the type of structural models, analysis methods, and computer codes used in the evaluations to extend the NI seismic analyses to soil sites. In the table, the applicant stated that the 2D finite element lumped-mass stick model of the ASB was analyzed using the SASSI Code, by time history analysis method for the purpose of parametric studies to establish the bounding generic soil conditions. However, during its review of the responses to other RAIs, the staff noted that 2D seismic analyses were apparently used for other purposes also. In RAI-TR03-34, the staff requested that the applicant clarify the information provided in Table 4.2.4-1, and update this table, as needed, to identify all applications of 2D seismic analysis, and how the results were used.

In its response dated July 5, 2007, the applicant stated that Table 4.2.4-1 had been revised to show the additional seismic models and analyses identified. The revision to the table also added the polar crane models and the CV shell model, included in the response to RAI-TR03-20. During the May 2008 audit, the staff verified that TR-03, Revision 1 included the revised Table 4.2.4-1, documenting the use of 2D analysis models. However, additional errors were found in the table. In a letter dated August 20, 2008, the applicant submitted a proposed revision to TR-03 Table 4.2.4-1. In a subsequent revision to TR-03, the applicant made appropriate changes to Table 4.2.4-1, which resolve this issue.

3.7.2.4.2 Nuclear Island Seismic Analysis using HRHF Input Motion

Subsequent to NUREG-1793 for the AP1000 DCD, Revision 15, the applicant added AP1000 DCD Appendix 3I in Revisions 16 and 17, in order to address the adequacy of the AP1000 seismic design for ground response spectra typical of CEUS HR sites, which are "rich in the high frequency range." These sites are referred to as HRHF sites. The applicant's technical basis for AP1000 DCD Appendix 3I is TR-115.

In May 2008, the staff issued ISG-1 on acceptable methods to demonstrate seismic adequacy for HRHF ground spectra. The four key elements of the guidance are:

- Use of the staff-accepted Abrahamson coherency function, to reduce the effects of the high-frequency ground motion.

- Use of a staff-accepted computer code (e.g., ACS SASSI) specifically developed to include the effects of incoherency.

- Use of building structural models sufficiently refined to adequately predict modal response up to 50 Hz.

- Selection of an adequate sampling of SSCs for detailed evaluation of response to the HRHF ground spectra.

The staff reviewed AP1000 DCD Appendix 3I and TR-115 using the elements of the ISG-1, in full consideration that the applicant's submittal represent the industry's first attempt to implement ISG-1.

The staff reviewed the introduction to TR-115, Revision 0, Section 1.0, and noted that the first paragraph stated that the purpose of the report is two-fold: (1) to confirm that high frequency seismic input is not damaging to equipment and structures qualified by analysis for the AP1000 CSDRS; and (2) to demonstrate that normal design practices result in an AP1000 design that is safer and more conservative than that which would result if designed for the high frequency input. The staff found that the above statements, made by the applicant, were too generic in nature, and required a qualification that they apply only to the HRHF spectra actually used in the analyses. The staff also noted that the last paragraph to the introduction section of TR-115 needed to be similarly qualified. In RAI-SRP3.7.1-SEB1-02, the staff requested that the applicant revise the stated purpose of TR-115, accordingly.

In a letter dated April 25, 2008, the applicant proposed changes to the introduction section of TR-115, to satisfy the staff's concern. The staff evaluated the RAI response and the proposed revisions to TR-115, and found them acceptable. The staff subsequently confirmed that TR-115, Revision 1, included the proposed revisions.

Although the applicant clarified the purpose of TR-115, the staff determined that the report contained insufficient information regarding site parameter requirements. The staff requested that the applicant specifically identify in TR-115 the minimum Vs of the underlying medium that must be satisfied in order to reference the results in TR-115, and also provide the technical basis for this determination. The staff noted that the definition of an HR site in the AP1000 DCD is a site with a minimum Vs of 2438.4 m/s (8,000 fps).

In a letter dated September 12, 2008, the applicant responded that the only requirement that COL applicants must demonstrate, to be covered by TR-115, is that their site GMRS is enveloped by the HRHF spectra. The applicant stated that sites with high Vs have higher loads due to a higher frequency than those with lower Vs, and sites that are enveloped by the HRHF input spectra, but have lower Vs, will have lower HRHF seismic loads than those used in the evaluation reported in TR-115.

The staff evaluated the supplemental response, and determined that the applicant's statement, that only a spectrum comparison is necessary, has no established technical basis. Softer material beneath the foundation will shift spectral peaks; whether the results for softer materials are enveloped by the HR results needs to be demonstrated. Based on the above assessment, the staff submitted Supplement 2 to RAI-SRP3.7.1-SEB1-02, requesting the applicant to address the following:

(a) Describe in detail the modeling of underlying media and any side media in the special SASSI analyses of the HRHF GMRS. How many cases were analyzed? Describe each case and the purpose for each case.

(b) What is the Vs associated with each of the media included in the SASSI analyses?

(c) How was the seismic motion at the surface developed for input to the SASSI analyses? Was the HRHF GMRS applied directly as surface motion, or was the surface motion developed from the HRHF GMRS applied at the NI foundation level? If the latter, describe in detail the method used to calculate the surface motion.

(d) Define numerically the range of Vs of the underlying media for which the special SASSI analyses are valid. Provide a detailed technical basis for this determination (e.g., results from parametric studies, previous documented studies, documented test results, "expert" judgment, etc.).

(e) For all COL applications that reference AP1000 DCD Appendix 3I and/or TR-115, are the site characteristics enveloped by the range of Vs defined in (d) above?

In a letter dated February 19, 2009, the applicant responded to RAI-SRP3.7.1-SEB1-02 (Supplement 2). The applicant presented a table of Vs versus depth for the single HRHF analysis conducted, but also restated its contention that only a spectral comparison is required. The staff found the applicant's response to Supplement 2 did not resolve the issue, and discussed this with the applicant in a teleconference on March 5, 2009. The applicant agreed that it is necessary for a specific site to satisfy both the response spectra criteria and also the Vs profile, in order to be covered by the analysis reported in TR-115.

In a letter dated April 14, 2009, the applicant revised its response to RAI-SRP3.7.1-SEB1-02 (Supplement 2), stating that either both requirements must be met, or a site-specific evaluation is needed. The applicant also identified a proposed revision to AP1000 DCD Sections 2.5.2.1 and 2.6, to incorporate this information. On the basis that the applicant has identified both essential requirements, the response to RAI-SRP3.7.1-SEB1-02 is acceptable. In a revised response dated July 9, 2010, the applicant indicated that a statement will be added to TR-115 that a comparison of the site-specific Vs profile to the generic HRHF Vs profile is needed in addition to the comparison of the site-specific spectra to the generic HRHF spectra. In subsequent revisions to the AP1000 DCD and to TR-115, the applicant made appropriate changes to the DCD and report text, which resolve this issue.

In a revised response dated July 9, 2010, the applicant indicated that a statement will be added to TR-115 to indicate that a comparison of the site-specific Vs profile to the generic HRHF Vs profile is needed in addition to the comparison of the site-specific spectra to the generic HRHF spectra. In a subsequent revision to the AP1000 DCD and TR-115, the applicant made appropriate changes to the DCD and report text, which resolve this issue.

The staff reviewed the description of ”Evaluation Methodology” in TR-115, Section 3.0, and noted that the methodology is consistent with the presentation made by the applicant during the April 2007 audit. However, TR-115, Section 3.0, does not include any of the quantitative information presented at the audit to demonstrate the implementation of the approach. In RAI-SRP3.7.1-SEB1-03, the staff requested that the applicant make available for audit, a detailed report of numerical results that demonstrate the implementation specifically for the AP1000. During the May 2008 audit, the staff reviewed the applicant’s report, which documents the implementation of the methodology, and concluded that it is consistent with the presentation made to the staff during the April 2007 audit and the staff’s ISG on incoherency. Initially, the staff considered RAI-SRP3.7.1-SEB1-03 to be resolved. However, the applicant subsequently revised the ACS-ANSYS NI20 model used for the HRHF analysis, in order to correct modeling errors identified by the staff during its confirmatory analysis effort. The staff identified the review of the revised analysis results as Open Item OI-SRP3.7.1-SEB1-03 in the SER with open items. During the June 14-18, 2010 audit, the staff reviewed the revised NI20 [ ] model (in calculation report, [ ]) to ensure that modeling corrections had been addressed. The staff verified that the SASSI model properly represented the actual AP1000 NI structural features. The staff also confirmed that seismic motion incoherency was implemented in accordance with the staff’s ISG. Based on the staff’s audit of [ ], RAI-SRP3.7.1-SEB1-03 and the associated open item are resolved.

The staff reviewed the details of the “Screening Criteria” in Section 4.0 of TR-115. The applicant lists four screening criteria used to select SSCs for detailed evaluation. Based on the screening criteria, it was not clear to the staff why the containment structure is not identified for detailed comparison of the CSDRS and the HRHF responses. In RAI-SRP3.7.1-SEB1-04, the staff requested that the applicant either include a detailed comparison for the containment structure in Section 6.1, or describe in detail its technical basis for excluding the containment structure.

In its response dated April 25, 2008, the applicant stated that the steel containment structure was not chosen for evaluation since it does not meet the criterion of significant modal response within the region of high frequency amplification. The applicant stated that the dominant frequencies for horizontal response are below 10 Hz, and the dominant mode in the vertical direction is below 20 Hz, which are not in the region where the HRHF spectra exceed the AP1000 CSDRS; and that over 75 percent of the containment structure mass participates in modes below the frequency where the HRHF spectra exceed the CSDRS. The staff evaluated the above response and initially concluded that the basis for excluding the containment shell was adequately described. However, the staff subsequently noted that AP1000 DCD Revisions 16 and 17, Section 3G.2.1.3, identifies high frequency modes (20-30 Hz) in the upper closure dome of the steel containment. Since high frequency modes in the upper closure dome were not addressed in TR-115, or in the initial RAI response, the staff requested that the applicant submit a supplemental RAI response justifying why these modes in the upper closure dome would not be excited by the HRHF ground spectra.

In its supplemental response dated September 12, 2008, the applicant stated that the seismic response spectra in the vicinity of the polar crane (~68 m (~224 ft) El.) are representative of the seismic response of the upper closure dome, and that the CSDRS spectra envelope exceeds the HRHF FRS at this location. Therefore, the applicant concluded that the closure dome will have lower response due to HRHF excitation than due to CSDRS excitation. The staff found this response to be inadequate because the results being compared are based on the stick model of the containment structure, which does not include the flexibility of the upper closure dome. The staff requested that the applicant provide information pertinent to addressing the staff's concern.

In its revised response dated May 14, 2009, the applicant stated that the NI20 ACS SASSI analysis for the HRHF ground motion input produced ISRS at the base of the SCV that are completely enveloped by the comparable ISRS produced by the CSDRS ground motion input, across the entire frequency range. The staff reviewed the comparison plots provided in the response, and noted that in this case the HRHF input would not excite the vibration modes in the SCV dome. The staff noted, however, that the applicant needed to confirm this after the HRHF reanalysis was completed. Pending the staff's evaluation of the applicant's revised incoherency analysis results (discussed under RAI-SRP3.7.1-SEB1-03, RAI-SRP3.7.1-SEB1-09, RAI-SRP3.7.1-SEB1-10, and RAI-SRP3.7.1-SEB1-11), this was designated as Open Item OI-SRP3.7.1-SEB1-04 in the SER with open items.

In its revised response dated July 9, 2010, the applicant provided updates to RAI response Figures RAI-SRP3.7.1-SEB-04-1, RAI-SRP3.7.1-SEB-04-2, RAI-SRP3.7.1-SEB-04-3, and RAI-SRP3.7.1-SEB-04-10 that show the corrected spectra comparisons. The staff noted that the CSDRS ISRS still envelope the HRHF ISRS, except for a very minor local exceedance in the Y direction ISRS. Since the input at the base of the SCV is more severe for the CSDRS than for the HRHF spectra, the staff accepts the applicant's decision to screen out the SCV from the HRHF detailed evaluation sample. RAI-SRP3.7.1-SEB1-04, and the associated open item, are resolved.

The staff's reviewed the analytical models described in TR-115, Section 5.0, and noted that the applicant had not adequately justified the applicability of the NI20 model to accurately predict high frequency modes potentially excited by the HRHF ground motion input. In RAI-SRP3.7.1-SEB1-06, the staff requested that the applicant include in Section 5.1 of TR-115 a comparison of frequencies and mode shapes between the more refined NI10 model and the NI20 model, to demonstrate the adequacy of the NI20 model to accurately predict high frequency modes.

In its response dated April 25, 2008, the applicant stated that at the December 20, 2007, meeting between the staff and industry related to the high frequency seismic events, it was agreed that a maximum analysis frequency of 50 Hz would be sufficient to transmit the high frequency response through the model. The applicant further stated that using the NI20 model (mesh size of 6.1 m (20 ft), and the shortest wavelength of 42.1 m (138 ft), there are close to 7 nodes per wavelength, to transmit the high frequency through the finite elements; and stated that it is not necessary to include in Section 5.1 a comparison of frequencies and mode shapes between the NI10 and NI20 models.

During the May 2008 audit, the staff noted that NUREG-0800 Section 3.7.2 (Revision 3, March 2007) identifies the staff's expectations for demonstrating adequacy of the element refinement to accurately simulate behavior at the highest frequency of interest, and requested that the applicant submit additional information to demonstrate the adequacy of the NI20 model. The applicant submitted supplemental responses in September 2008, January 2009, and June 2009. The staff reviewed these supplemental responses and concluded that none of the information submitted directly addressed the staff's initial RAI question.

As a result of the inadequate responses from the applicant, the staff initiated an independent confirmatory analysis effort in June 2009, to study the modal properties of both the NI10 and NI20 models and compare the two models up to 50 Hz. Based on this effort, the staff concluded that the overall building response is adequately represented in the NI20 model, up to 50 Hz. However, local panel vibration modes of walls, floors, and ceilings, up to 50 Hz, are not necessarily modeled with sufficient refinement in the NI20 model. The staff's concern is that, if the NI20 model cannot accurately predict the amplified response of flexible regions up to 50 Hz, then any HRHF high frequency exceedances of the design ISRS (based on the CSDRS) cannot be accurately predicted. Therefore, the staff requested that the applicant review the NI20 model to determine which wall, floor, and ceiling panels are not modeled with sufficient refinement, and to address how this affects the structural design loads and the ISRS, for the HRHF ground spectra input. This was identified as Open Item OI-SRP3.7.1-SEB1-06 in the SER with open items.

In its revised response dated July 27, 2010, the applicant stated that the procedure for addressing the out-of-plane response of flexible regions was the same as that described in its revised response (July 9, 2010) to RAI-TR03-032. The applicant used the NI05 model to identify flexible regions where the out-of-plane response may occur at frequencies less than 50 Hz. The staff's review of the applicant's July 9, 2010 response RAI-TR03-032 is in Section 3.7.2.4.1 of this SER. Based on its review, the staff finds the applicant's method for identifying flexible regions (below 50 Hz) and modifying the ISRS to be acceptable. By using the mesh refinement of the NI05 model, the applicant was able to locate and evaluate flexible regions of the NI structures that were inadequately modeled in the less refined NI20 model. The applicant identified proposed changes to TR-115 to document the new procedure. Therefore, RAI-SRP3.7.1-SEB1-06 and the associated open item are resolved. In a subsequent revision to TR-115, the applicant made appropriate changes to the report text, which resolves this issue.

In its review of the NI10 and NI20 spectral comparisons in Section 5.1, the staff noted that the locations presented showed no significant amplification in the high frequency range. In RAI-SRP3.7.1-SEB1-08, the staff requested that the applicant include in Section 5.1, NI10 versus NI20 comparisons at locations and in directions where there is significant amplification at high frequency.

In its response dated September 12, 2008, the applicant stated that Figures 5.1-4 and 5.1-5 would be added to Section 5.1 of TR-115 to show the locations and response spectra at additional locations. The staff reviewed the supplemental response and found that the two added locations exhibit more significant response in the high frequency region than the three original locations. Significant spectral amplification in X and Y is generally in the 10-20 Hz range, with one Y-direction peak in the 20-30 Hz range. Significant spectral amplification in Z-direction is generally in the 20-30 Hz range.

The staff noted, however, that the comparisons presented did not demonstrate any consistent pattern of correlation among the three models (ANSYS NI10, ANSYS NI20, and SASSI NI20).

In two of the horizontal comparisons, there are significant differences in the 7-8 Hz range, where excellent correlation would be expected. The staff concluded that although the applicant's response addressed the information request, there was no discussion of the anomalies in the comparisons. The staff was concerned that the applicant had not conducted a sufficient assessment of these results before submitting them. Therefore, the staff issued RAI-SRP3.7.1-SEB1-08, Supplement 1, describing the anomalies and requesting the applicant to review and comment on them.

In it supplemental response dated February 24, 2009, the applicant stated that the results presented were obtained from different models (NI10 and NI20) and different technologies (ANSYS - time domain solution, and SASSI - frequency domain solution), and that this can result in the differences identified. The applicant stated that the response spectra show:

- In general the shapes of the response spectra are similar.
- The NI20 model has higher response than the NI10 model.
- SASSI analyses are conservative.

The staff determined that the applicant had not addressed the specific questions posed by the staff, and discussed this with the applicant during the April 2009 audit. The applicant agreed to provide additional information to explain the inconsistencies noted by the staff.

In a letter dated June 3, 2009, the applicant submitted a supplemental response to this RAI, explaining that the inconsistent results reported in the Z direction between nodes 2247 and 2078 was due to modeling differences between the NI10 and NI20 models. The staff reviewed the additional information, and concluded that the explanation is plausible, but not conclusive. The staff determined that resolution of this RAI would need to be deferred until the staff had completed its independent confirmatory analysis program. This was identified as Open Item OI-SRP3.7.1-SEB1-08 in the SER with open items.

The results of the staff's confirmatory analysis of the NI20 SASSI model are described under OI-SRP3.7.1-SEB1-09, OI-SRP3.7.1-SEB1-10, and OI-SRP3.7.1-SEB1-11. The staff identified errors in the applicant's NI20 SASSI model, which required the applicant to perform a reanalysis of all SASSI runs. During the June 14-18, 2010 audit, the staff reviewed the revised NI20 SASSI model and results (in calculation report, [ ]). The staff verified that the revised SASSI model properly represented the actual AP1000 NI structural features.

In its revised response dated July 9, 2010, the applicant indicated that the differences in response between the southeast and northeast corners of the auxiliary building, as depicted in corrected TR-115 Figures 5.1-7 and 5.1-8, are due to local differences in geometry between the NI10 and NI20 models, and also due to differences in the seismic ISRS at the base of the auxiliary building, between SASSI and ANSYS. The applicant also identified a proposed revision to TR-115. The staff determined that the applicant's response is acceptable, on the basis that these results are not design-basis results, but are only intended to demonstrate dynamic similarity between the three models (ANSYS NI10, ANSYS NI20, and SASSI NI20). Also, as discussed under RAI-SRP3.7.1-SEB1-06, there are local regions where NI20 does not possess the necessary model refinement to represent modal behavior up to 50 Hz. In these areas, the applicant is relying on the more refined NI05 model to develop HRHF ISRS. Therefore, the staff considers RAI-SRP3.7.1-SEB1-08, and the associated open item to be resolved. In a subsequent revision to TR-115, the applicant made appropriate changes to the report text, which resolves this issue.

The staff reviewed the HRHF ISRS presented in TR-115, Section 5, and issued three related RAIs. RAI-SRP3.7.1-SEB1-09, RAI-SRP3.7.1-SEB1-10, and RAI-SRP3.7.1-SEB1-11 requested that the applicant provide clarification and explanation of in-structure response reductions and apparent inconsistencies in the presented results. The significant issues raised by the staff and the applicant's responses follow.

(1) The staff noted that the spectral acceleration ratio of coherent motion to incoherent motion is as high as 3, and a ratio of 2 is fairly common. The staff requested that the applicant provide the detailed technical basis for concluding that the calculated reductions are reasonable, and consistent with the ISG on this subject; and also to identify whether any independent peer review of this result had been performed, considering it is a first-time application of this technology.

(2) The staff noted that spectral acceleration reductions are indicated at frequencies as low as 6-10 Hz. The staff requested that the applicant provide the detailed technical basis for concluding that the calculated reductions at a low frequency are reasonable, and consistent with the ISG on this subject; and also to identify whether any independent peer review of this result had been performed, considering it is a first-time application of this technology.

(3) The staff noted that even when the beneficial effects of incoherency are included, there are high frequency exceedances at a number of the sample locations evaluated. However, the applicant apparently has concluded that the worst-case exceedances have been determined, without expanding the sample size and evaluating additional locations. The staff requested that the applicant provide a detailed technical basis for concluding that the seismic response of AP1000 SSCs to the defined HRHF ground spectra input is enveloped by the response at the selected sample locations.

(4) The staff reviewed the ISRS for the containment operating floor, east side, El. 40.9 m (134.25 ft) (Node 2136), and for the containment operating floor, west side, El. 40.9 m (134.25 ft) (Node 2170), in TR-115, Revision 1, Figure 5.2-2. The staff observed that the east side and west side Y-direction spectra are very similar. However, the east side and west side X-direction spectra and the east side and west side Z-direction spectra are very different, for both the HRHF-coherent and HRHF-incoherent cases.

Location	Direction	HRHF-coherent	HRHF-incoherent
East Side	X	1.6g (20 Hz)	1.05g (20 Hz)
West Side	X	3.5g (13 Hz)	2.8g (13 Hz)
East Side	Y	3.5g (16 Hz)	1.95g (16 Hz)
West Side	Y	3.7g (16 Hz)	2.05g (16 Hz)
East Side	Z	1.9g (40-50 Hz)	0.65g (40-50 Hz)
West Side	Z	3.2g (30 Hz)	1.7g (30 Hz)

The staff could not determine a rational explanation for this behavior, and requested that the applicant provide a detailed technical explanation for these apparently inconsistent results.

In a letter dated February 4, 2009, the applicant provided the following response:

(1) SASSI-Simulation incoherency approach used to generate the seismic response spectra is in accordance with Section 4, Section 1.0 of "Interim staff Guidance (ISG) on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combine License Applications," supplements to Section 3.7.1, "Seismic Design Parameters," of NUREG-0800. In generating the seismic response spectra, the applicant made no changes to the accepted industry methodology. The technical basis for incoherence is discussed in EPRI Report 1012966, "Effect of Seismic Wave Incoherence on Foundation and Building Response," December 2005. Similar results were shown in Figure 6-1 to 6-11 of EPRI Report 1012966. Figure 6-12 showed 5-fold reduction at 50 Hz.

(2) See (1) above. Figure 6-6 of EPRI Report 1012966 showed the similar reduction at 10 Hz.

(3) The applicant had agreed to evaluate a representative sample of SSCs located in areas that are subject to high frequency response, and have frequency content in the high frequency region, to confirm that high frequency seismic input is not damaging, and to demonstrate that normal design practices using the CSDRS result in an AP1000 design that is safer and more conservative. This evaluation is reported in TR-115. The SSCs selected based on the screening criteria are sufficient to demonstrate that high frequency seismic events are not damaging. There may be spectra that have higher exceedances; however, safety-related equipment may not be located in these locations, SSCs located in these areas may not have high frequency response, and further the evaluation performed demonstrates that the HRHF seismic event is not damaging and there is margin between the CSDRS and HRHF response. The applicant's evaluation approach is in compliance with Section 4, Subsections 3.0 and 4.0 of the "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications."

(4) Figure RAI-SRP3.7.1-09-C (in the response) shows the location of nodes 2136 and 2170. Node 2170 is surrounded by a large semi-circle IRWST water tank while node 2136 is surrounded by concrete structure floor and steam generator compartment wall. Node 2136 showed more interaction in X and Z direction between the CISs. The responses of both nodes in Y direction are similar because of less structure interaction between the steam generator compartment wall and other concrete structure. The differences between coherent and incoherent responses are justified in (1) and (2) above.

The staff reviewed the applicant's responses to the supplemental information request, and determined that the responses to (1) and (2) were unacceptable, because the applicant referenced an EPRI report that is not referenced in ISG-1. The applicant needed to confirm that it used the specific reports referenced in the ISG, dated May 19, 2008. If this is not the case, then the applicant would need to perform new analyses that are consistent with the ISG approved methods. Also, the applicant had to confirm that the results questioned by the staff in (1) and (2) are consistent with results presented in TRs that the staff has accepted.

The staff discussed this RAI response with the applicant during the April 2009 audit. The staff determined that the best course of action to resolve the remaining staff concerns on Items (3) and (4) was to conduct independent confirmatory analyses. To support this effort, the applicant agreed to submit the SASSI NI20 model used in its incoherency analyses to the staff. The staff also requested that the applicant conduct several parametric analyses, using a simplified AP1000 model from the EPRI studies and varying the basemat dimensions and properties of the foundation media.

Confirmatory Analysis

To support the staff's review of the applicant's responses to RAI-SRP3.7.1-SEB1-09, RAI-SRP3.7.1-SEB1-10, and RAI-SRP3.7.1-SEB1-11, the staff initiated a confirmatory analysis effort in May 2009. The applicant provided the staff with the seismic analysis models (ANSYS NI20 and SASSI NI20), so that an independent check of modeling assumptions could be performed. In the confirmatory analysis effort, the staff identified several key findings:

1. The staff identified several modeling errors in the applicant's SASSI NI20 model. The errors related to the end-release assumptions for certain beam elements and their effect on over-constraining the global SASSI model. In addition, there were several foundation nodes on the NI basemat that were not identified as SASSI interface nodes. It was not clear to what extent these modeling errors might affect ISRS as well as the ZPA values used for structural design. The staff informed the applicant, during the August 2009, audit in Cranberry, Pennsylvania, of these errors and that the errors are likely to affect the results presented in TR-115 and TR-03. The applicant agreed to submit revised results for all prior SASSI analyses reported in TR-115 and TR-03.

2. The staff studied the adequacy of the NI20 model refinement to reasonably predict all vibration modes up to 50 Hz, as specified in the ISG. The conclusion is that there are local regions (i.e., floor, wall, and roof slabs) where the refinement is not sufficient to pick up a local 50 Hz vibration mode. Therefore, the ISRS may not be accurate in these areas. In RAI-TR03-032 and RAI-SRP3.7.1-SEB1-06, the staff requested that the applicant review the NI20 model, locate all such local areas, determine whether there are mounted systems and components in these areas, and describe how the appropriate ISRS will be developed for these areas.

3. The staff compared results between ACS SASSI and the latest version of SASSI 2000, for the AP1000 NI20 model and HRHF ground motion, with and without incoherency effects. There are significant reductions in the low frequency region of the ISRS when incoherency effects are included. The staff found that the low frequency reductions were not consistent with EPRI calculations referenced in ISG-1. The staff's review of the applicant's use of incoherency is discussed below.

Use of Incoherency

The staff focused its review of the applicant's use of spatial incoherency by requesting the applicant (RAI-SRP3.7.1-SEB1-10) to provide comparisons of ISRS using both coherent and incoherent input motion. In response to RAI-SRP3.7.1-SEB1-10, the applicant provided response spectra comparisons at several locations on the NI:

A. Top of the shield building (El. 99.8 m (327.4 ft))
B. East side of the containment operating floor (El. 40.9 m (134.25 ft))
C. West side of the containment operating floor (El. 40.9 m (134.25 ft))
D. Shield building, northeast corner (El. 40.9 m (134.5 ft))

E. Shield building, at fuel building roof (El. 54.7 m (179.6 ft))
F. Reactor coolant pump (RCP) (El. 30.2 m (99.0 ft))

For the purpose of comparing the applicant's results to previous EPRI calculations, the staff reviewed the response spectra comparisons, and developed approximate ratios of incoherent to coherent motion in the low and high frequency ranges. These comparisons are provided in SER Table 3.7-1. The applicant also stated that the Abrahamson Hard-Rock Coherency Model (2007), as incorporated into ACS- SASSI, was used to perform SSI calculations. The staff finds that the applicant's use of the 2007 Abrahamson Hard-Rock coherency model is consistent with staff guidance (i.e., ISG-1).

Table 3.7-1. Incoherent Versus Coherent Response (Approximate)

Building Location	Direction	Incoherent/Coherent Response Ratio	
		0-10 Hz	10-50 Hz
Top of the shield building	X	0.90	0.75
	Y	0.95	0.85
	Z	0.65	0.90
East side of the containment operating floor	X	0.90	0.75
	Y	0.90	0.70
	Z	0.90	0.55
West side of the containment operating floor	X	0.90	0.85
	Y	0.85	0.75
	Z	0.90	0.50
Shield building, Northeast corner	X	0.85	0.70
	Y	0.95	0.75
	Z	0.80	0.65
Shield building, at fuel building roof	X	0.85	0.75
	Y	0.80	0.75
	Z	0.80	0.60
Reactor coolant pump	X	0.90	0.90
	Y	0.80	0.95
	Z	0.75	0.85

The results shown in SER Table 3.7-1 indicate that low frequency reductions range from 5-35 percent. The locations of the most significant response reductions are at the top of the shield building and at the RCP, with approximately 25-35 percent reductions in the 0-10 Hz range.

High-frequency response reductions range from 5-50 percent. The locations of the most significant high-frequency reductions are at the east and west sides of the containment operating floor, in the vertical direction, and the shield building (at fuel building roof), in the Y direction. Approximate reduction of 45-50 percent in the 10-50 Hz range was observed at these locations.

The staff also reviewed spectral response comparisons for several nodes on the basemat. These basemat nodes exhibited similar reductions in response both in the low and high frequency ranges. The staff finds that the high-frequency response predictions are reasonable based on comparisons with similar calculations performed by EPRI (TR-1015111, 2007) using more simplified structural models. However, the staff finds that the applicant's low-frequency response reductions, in excess of 30 percent, to be unsupported by the EPRI calculations. To address this concern, in RAI-SRP3.7.1-SEB1-11, the staff requested that the applicant provide justification for the significant reductions in a low frequency response.

In its response, the applicant stated that the low frequency reductions were due to the use of the 2007, HR coherency function itself, which can have a 50 percent reduction at 50 m (164 ft) in the 2-5 Hz range. The staff found the applicant's justification inadequate because the applicant referenced EPRI calculations (TR-1015111, 2007, Chapter 5), which are based on a soil coherency model that is not applicable to HR sites. The staff notes that Appendix B of the same EPRI report includes results using the approved 2007 coherency function and serves as the staff's basis for comparison.

The staff investigated the applicant's low-frequency response predictions. With the intent of reducing computational effort, the staff developed a simplified FEM of the AP1000 NI. This reduced model was then used for SSI analysis using the ACS- SASSI and SASSI -square root of the sum of the square (SRSS) codes. The simplified SSI model had dynamic response characteristics similar to those of the applicant's more detailed NI model, for frequencies below about 15 Hz. The dynamic response of the simplified model was confirmed by comparing fixed-base TFs at several locations to the more detailed AP1000 NI model. A transfer function is defined as a frequency-dependent function of SSI amplification due to a unit input motion. Further, for incoherent analysis using both analytic formulations recognized by the ISG, the confirmatory analyses used the same 2007 Abrahamson coherency function that the applicant referenced, as well as the applicant's HRHF input motion.

The staff performed SSI analyses using the simplified model for both coherent and incoherent motion. The goal of this analysis was to determine if the low frequency reductions of ISRS seen in the applicant's analysis could be duplicated with SASSI-SRSS. This analysis also used the same HR site and HRHF input motion provided by the applicant.

The SSI analysis results using SASSI -SRSS for the simple NI model, as well as the full NI20 FEM with HRHF input, indicate negligible reductions in ISRS in the low frequency range due to incoherency effects. SSI TFs of the simplified model from both SASSI -SRSS and ACS- SASSI show negligible reductions in the low frequency range (below 10 Hz). In addition, 5 percent damped ISRS from SASSI -SRSS analysis of the NI20 model exhibit only negligible reductions at low frequency.

Based on the review of the applicant's results and the staff's independent confirmatory analysis efforts, the staff concluded that the applicant's predictions of in-structure response in the low frequency range were not consistent with EPRI's calculations and the staff's confirmatory calculations. The staff also noted that the applicant's high-frequency incoherent results cannot be considered acceptable if low frequency results cannot be validated. These issues are identified as Open Items OI-SRP3.7.1-SEB1-09, OI-SRP3.7.1-SEB1-10, and OI-SRP3.7.1-SEB1-11.

During the June 14-18, 2010 audit, staff reviewed the revised NI20 SASSI model (in calculation report, [ ]) to ensure that modeling corrections had been addressed. The staff verified that the SASSI model was properly transferring bending moments at the beam (or shell) connections with solid elements. In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-SRP3.7.1-SEB1-11. The applicant indicated that [ ] modeling corrections (e.g., beam element and shell element connections to solid elements) had been addressed and the reanalysis had been performed.

The applicant provided ACS SASSI results for the corrected NI20 model. Using the incoherency option in ACS SASSI, the applicant developed ISRS results for 25 simulations (with and without phase adjustment) for the AP1000 NI six key locations (shown in Figures RAI-SRP3.7.1-SEB1-11-50 through RAI-SRP3.7.1-SEB1-11-67 of the response). The staff reviewed these comparisons and finds that while there are some differences between the original HRHF results and the corrected results (with phase adjustment), the original HRHF results are generally conservative.

The applicant also provided ISRS comparisons (coherent and incoherent) at the four corners and center of the NI basemat (shown in Figures RAI-SRP3.7.1-SEB1-11-68 through RAI-SRP3.7.1-SEB1-11-82 of the response). The applicant stated that these analyses incorporate a phasing correction, which no longer results in significant low-frequency reductions. The staff reviewed these comparisons and finds that there are minimal (<10 percent) ISRS reductions below 10 Hz for the locations presented.

Based on review of the applicant's corrected NI20 SASSI model and the new HRHF results, the staff finds that the applicant has properly implemented modeling corrections, and the ISRS show negligible reductions due to incoherency below 10 Hz. On the basis of these findings, RAI-SRP3.7.1-SEB1-11 and associated open item are resolved.

In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-SRP3.7.1-SEB1-09. In response to a request from the staff, the applicant identified the following proposed addition to TR-115, Section 5.2:

> The exceedances of CSDRS-based ISRS by HRHF-based ISRS are addressed as part of the sampling evaluation documented in this report to confirm that high frequency input has marginal effect on equivalent piping, and structures qualified by analysis for the AP1000 CSDRS.

The applicant had previously addressed issue (4) described above in its February 9, 2009, response, by providing Figure RAI-SRP3.7.1-09-C in the response, which shows the location of nodes 2136 and 2170, and stated that node 2170 is surrounded by a large semi-circular IRWST water tank, while node 2136 is surrounded by concrete structure floor and steam generator compartment wall. The applicant noted that node 2136 showed more interaction in X and Z direction between the CISs. The responses of both nodes in Y direction are similar because of less structure interaction between the steam generator compartment wall and other concrete structure. Prior to the staff's confirmatory analysis, and the applicant's reanalysis after correction of modeling errors, the staff had reserved judgment on the applicant's explanation. With the resolution of RAI-SRP3.7.1-SEB1-10 and RAI-SRP3.7.1-SEB1-11, the staff has concluded that the applicant's explanation for the differences is viable. Therefore, RAI-SRP3.7.1-SEB1-09, and the associated open item, are resolved. In a subsequent revision to TR-115, the applicant made appropriate changes to the report text, which resolves this issue.

Acceptability of ISRS Reductions

In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-SRP3.7.1-SEB1-10, which provided the reanalysis for seismic response, using the corrected NI20 model. In Figures RAI-SRP3.7.1-SEB1-10-1 to RAI-SRP3.7.1-SEB1-10-21 of the response, the applicant provided incoherent and coherent ISRS comparisons. The applicant stated that some ratios of incoherent-to-coherent response are shown to be less than 0.5. To justify this level of reduction, the applicant used the EPRI AP1000 stick model to compare ISRS reductions to the 3D AP1000 model. Three cases were analyzed: EPRI stick model with EPRI soil profile and EPRI time history; EPRI stick model with EPRI soil profile and HRHF time history; EPRI stick model with HRHF soil profile; and HRHF time history input. The results of these analyses are shown in Figures RAI-SRP3.7.1-SEB1-10-22 to RAI-SRP3.7.1-SEB1-10-33 of the response. The results showed that a larger foundation will have a larger reduction in response due to incoherency effects. The results for the top-of-CIS show reductions of the magnitude seen in the NI20 results (approximately 50 percent reduction). The top of the SCV and top the shield building also show similar results. Figures RAI-SRP3.7.1-SEB1-10-34 and RAI-SRP3.7.1-SEB1-10-35 of the response show a comparison of the basemat response of the NI20 model and the EPRI stick models. The comparison shows that the reductions due to incoherency are similar in magnitude.

The staff reviewed the applicant’s comparison of incoherent and coherent results and finds the results similar to those developed independently (SER Table 3.7-1). Based on the similar ISRS reductions of the AP1000 to the EPRI calculations (which are referenced in the ISG), the staff finds the applicant’s reductions due to the use of incoherency to be acceptable. Therefore, RAI-SRP3.7.1-SEB1-10, and the associated open item are resolved.

Evaluation of Structures for HRHF Loading

During the April 2007 audit, the applicant presented structural response comparisons between CSDRS loading and HRHF loading. The staff obtained clarification from the applicant that the HRHF results assumed coherent motion. However, the staff noted that TR-115, Section 6.1, did not identify whether the structural response comparisons in Tables 6.1-1 through 6.1-6, between CSDRS loading and HRHF loading, assumed coherent motion or incoherent motion. In RAI-SRP3.7.1-SEB1-12, the staff requested that the applicant clearly define how it calculated the HRHF structural loads presented in TR-115, Tables 6.1-1 through 6.1-6.

In a letter dated April 25, 2008, the applicant stated that the HRHF member forces provided in Tables 6.1-1 through 6.1-6 are based on incoherency. The incoherent member forces are averaged from 25 independent Monte Carlo runs done with [ ] and multiplied by the element thickness to form the member forces presented.

The staff also requested, in RAI-SRP3.7.1-SEB1-13, that the applicant provide additional comparison results in Tables 6.1-1 through 6.1-6, based on use of the HRHF ground motion without considering reduction for incoherency, similar to the results presented in April 2007. In a letter dated April 25, 2008, the applicant provided the requested comparisons between the coherent and incoherent results in a set of tables designated RAI-SRP3.7.1-SEB1-13-01 to RAI-SRP3.7.1-SEB1-13-01-6. The applicant also noted that it had identified inconsistencies in the HRHF incoherent results tabulated in TR-115, and referred to its response to RAI-SRP3.7.1-SEB1-14.

During review of TR-115, Tables 6.1-1 through 6.1-6, the staff had noted several erratic patterns of differences between the CSDRS results and the HRHF results. In RAI-SRP3.7.1-SEB1-14, the staff requested that the applicant review the tabulated results in Tables 6.1-1 through 6.1-6, and provide a technical explanation for all patterns of differences that the applicant determined to be in need of further review.

In a letter dated April 25, 2008, the applicant stated that it had reviewed the tabulated results in Tables 6.1-1 through 6.1-6 and concluded that there were inconsistencies in the tabulated results. These inconsistencies were corrected; the revised tables were included in the RAI response, and also identified for inclusion in TR-115, Revision 1. The applicant stated that the conclusions in Section 6.1 remain unchanged. During the May 2008, audit, the staff discussed these three RAI responses with the applicant. The expanded and corrected results included in the response to RAI-SRP3.7.1-SEB1-13 show that the HRHF coherent results are enveloped by the CSDRS results. Therefore, the staff concluded that structures designed to the CSDRS input are also adequately designed for the HRHF input. The staff also confirmed that the corrected tables were included in TR-115, Revision 1. On this basis, RAI-SRP3.7.1-SEB1-12, RAI-SRP3.7.1-SEB1-13, and RAI-SRP3.7.1-SEB1-14 are resolved.

3.7.2.5 Development of Floor Response Spectra

In AP1000 DCD, Revision 19, Section 3.7.2.5, the applicant stated that design FRS are generated according to RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” Revision 1. The seismic FRS is computed using time-history responses determined from the NI seismic analyses. The time-history responses for the HR condition are determined from a mode superposition time history analysis using computer program [ ]. The time-history responses for the FR and soil conditions are determined from a complex frequency response analysis using the computer program, [ ]. FRS for damping values equal to 2, 3, 4, 5, 7, 10, and 20 percent of critical damping are computed at the required locations.

The applicant stated that FRS for the design of subsystems and components are generated by broadening the enveloped nodal response spectra determined for the HR site and soil sites. The spectral peaks are broadened by ±15 percent to account for the variation in the structural frequencies, due to the uncertainties in parameters, such as material and mass properties of the structure and soil, damping values, seismic analysis technique, and the seismic modeling technique. Figure 3.7.2-14 shows the broadening procedure used to generate the design FRS.

The applicant further stated that spectral peaks at frequencies associated with fundamental SSI frequencies are reviewed. If there is a “valley” between peaks due to different soil profiles and not the building modal response, then this valley is filled by extending the broadening of the lower peak horizontally until it meets the broadened upper peak. The SSE FRS for 5 percent damping, at representative locations of the coupled ASBs, the SCV, and the CIS, are presented in AP1000 DCD, Revision 19, Appendix 3G.

Based on its review of AP1000 DCD, Revision 19, Section 3.7.2.5, and the related information in Appendix 3G, the staff concluded that the applicant’s approach for enveloping the multiple site responses, and filling any “valley” in the envelope attributable to soil response, is consistent with current staff guidance, and is acceptable.

3.7.2.6 Three Components of Earthquake Motion

In AP1000 DCD Section 3.7.2.6, the applicant stated that seismic system analyses are performed considering the simultaneous occurrences of the two horizontal and the vertical components of earthquake. In mode superposition time-history analyses using the computer program, ANSYS, the three components of earthquakes motions are applied either simultaneously or separately. In the ANSYS analyses with three component earthquake motion applied simultaneously, the effect of the three components of earthquake motion is included within the analytical procedure so that further combination is not necessary. In analyses where the earthquake components are applied separately, the three components of earthquake motion are combined using one of the following methods:

- For seismic analyses with the statistically independent earthquake components applied separately, the time-history responses from the three earthquake components are combined algebraically at each time step to obtain the combined response time-history. This method is used in the SASSI analyses.

- The peak responses due to the three earthquake components from the response spectrum and equivalent static analyses are combined using the SRSS method.

- The peak responses due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak (100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is used in the NI basemat analyses, the CV analyses and the shield building roof analyses.

The applicant further stated that the CV is analyzed using axisymmetric FEMs. These axisymmetric building structures are analyzed for one horizontal seismic input from any horizontal direction and one vertical earthquake component. Responses are combined by either the SRSS method or by a modified 100 percent-40 percent-40 percent method in which one component is taken at 100 percent of its maximum value and the other is taken at 40 percent of its maximum value.

The applicant stated that a summary of the dynamic analyses performed and the combination techniques used is presented in AP1000 DCD Appendix 3G. In Appendix 3G.4.3.1, the applicant indicated that for RSA, the SRSS method is used to combine the spatial components, in accordance with Section 2.1 of RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Revision 2.

The staff reviewed the update to AP1000 DCD Section 3.7.2.6, and related information in Appendix 3G, and concluded that: (1) algebraic combination at each time step is consistent with standard practice and the staff guidance for time history analyses using three statistically independent inputs, including SSI analyses using ANSYS, and is acceptable; and (2) use of the SRSS combination is consistent with standard practice and the staff guidance for RSA, equivalent static analysis, and time history analysis when the three inputs are not statistically independent, and is acceptable.

In NUREG-1793 for the AP1000 DCD, Revision 15, the staff had accepted the use of the 100-40-40 method for combining the responses due to the three components of earthquake motion, when the equivalent static acceleration method is used. In July 2006, the staff issued RG 1.92, Revision 2, which included guidance on implementation of the 100-40-40 method. After the submittal of the AP1000 DCD, Revision 17, the applicant identified significant design changes to the roof of the shield building, which is analyzed for seismic response using equivalent static analysis and the 100-40-40 combination method. In addition, equivalent static analysis and the 100-40-40 combination method are used for seismic evaluation of the containment structure and the basemat. Therefore, the staff inquired whether the applicant had implemented the 100-40-40 method in accordance with the guidance provided in RG 1.92, Revision 2. The staff's safety concern was that improper implementation of the 100-40-40 combination method may result in unconservative estimates of seismic demands. This issue was addressed by Open Item OI-TR85-SEB1-27. This open item has been resolved, and the staff has accepted the applicant's implementation of the 100-40-40 method, based on comparison of the applicant's results to results using the SRSS combination method. See Section 3.8.4.1.1.3.4 of this report for the staff's detailed assessment.

3.7.2.7 Combination of Modal Responses

In AP1000 DCD, Revision 17, Section 3.7.2.7, the applicant stated that the modal responses in a RSA are combined using the grouping method shown in Section C of RG 1.92, Revision 1, and when high frequency effects are significant, they are included using the procedure given in Appendix A to NUREG-0800 Section 3.7.2. The applicant further stated that in the fixed base mode superposition time history analysis of the HR site, the total seismic response is obtained by superposing the modal responses within the analytical procedure so that further combination is not necessary. This is unchanged from the AP1000 DCD, Revision 15.

A summary of the dynamic analyses performed and the combination methods used are presented in AP1000 DCD, Revision 17, Appendix 3G. In paragraph 3G.4.3.1, the applicant indicated that the RSA is conducted in accordance with Sections 1.1.3, 1.3.2, 1.4.2, and 1.5.2 of RG 1.92, Revision 2. The staff noted that the applicant's use of the guidance in RG 1.92, Revision 2, for combination of modal responses in RSA, is acceptable because it is consistent with the latest staff guidance on this subject.

However, the staff could not determine whether the applicant's mode superposition time history analyses adequately account for the residual rigid response associated with natural vibration modes with frequencies higher than the input spectrum ZPA frequency. RG 1.92, Revision 2, incorporates more recent research findings with respect to modal response combination methods and the treatment of residual rigid response. It is important to accurately account for the residual rigid response if a nuclear power plant SSC has significant natural vibration modes with frequencies higher than the input spectrum ZPA frequency. Ignoring the residual rigid response in these cases may result in significant underestimation of SSC element forces and moments in the vicinity of supports, as well as underestimation of support forces and moments. In RAI-SRP3.7.1-SEB1-17, part (d), the staff requested that the applicant identify whether the method employed is consistent with or different from the RG 1.92, Revision 2, approach, and to provide the technical basis for the adequacy of any method used that differs from the current staff guidance. The applicant's initial response to the staff's RAI was unsatisfactory. This was identified as Open Item OI-SRP3.7.1-SEB1-17 in the SER with open items.

In its revised response to RAI-SRP3.7.1-SEB1-17, part (d), dated July 27, 2010, the applicant stated that modal superposition time history analysis provides sufficient solution accuracy, without including the residual rigid response, because the modes, which respond beyond the ZPA frequency of the input have no significant contribution to the amplified ISRS. In order to verify the accuracy of the analyses conducted, the applicant performed time history analysis using the NI10 model, with a cutoff frequency of 44 Hz, and an identical time history analysis with additional modes up to 64 Hz for the ASB, and additional modes up to 100 Hz for the CIS. The ISRS comparisons at 5 percent damping are documented in the RAI response at key locations of the ASB and CIS. The applicant provided similar comparisons for key locations in the ASB NI05 model, for 40 Hz and 85 Hz cutoff frequencies. The staff reviewed the comparisons of the ISRS, which showed negligible differences in results between the 2 selected cutoff frequencies. These results support the applicant's position; therefore, the staff concluded that the applicant's implementation of the mode superposition time history analysis method produced sufficiently accurate results, even though it does not formally account for the residual rigid response above the cutoff frequency, as specified in RG 1.92, Revision 2. Therefore, RAI-SRP3.7.1-SEB1-17, part (d), and the associated open item are resolved.

3.7.2.8 Interaction of Noncategory I Structures With Seismic Category I Structures

In AP1000 DCD, Revision 17, Section 3.7.2.8, the applicant described the approach for evaluating the effects of interactions of noncategory I structures with seismic Category I SSCs, and components. The approach identified in the AP1000 DCD, Revision 15, remains unchanged. The evaluation must satisfy one of the following three criteria: (1) collapse of the noncategory I structure will not cause an impact with any seismic Category I SSC; (2) collapse of the noncategory I structure will not impair the intended function of any seismic Category I SSC; or (3) the noncategory I structure is classified as seismic Category II and is analyzed and designed to prevent its collapse under the SSE. The applicant identified three structures adjacent to the AP1000 NI: the annex building, the radwaste building, and the turbine building. There is no change between the AP1000 DCD, Revisions 15 and 17 for the radwaste building. The applicant's evaluation for the radwaste building was previously accepted by the staff.

In the AP1000 DCD, Revision 17, the applicant revised the seismic classification of the annex building. In AP1000 DCD, Revision 15, the entire annex building was classified as seismic Category II. In AP1000 DCD, Revision 17, Section 3.7.2.8.1, the applicant stated that only the portion of the annex building adjacent to the NI is classified as seismic Category II. The applicant stated that the annex building is analyzed for the SSE for the six soil profiles described in AP1000 DCD Section 3.7.1.4 and that for the HR site, a range of soil properties was assumed for the layer above rock at the level of the NI foundation. In RAI-SRP3.7.1-SEB1-15, part (b), the staff requested that the applicant clarify the seismic classification of the remainder of the annex building and confirm that for analysis purposes, the entire annex building has been treated as seismic Category II.

In its initial response dated February 6, 2009, the applicant stated that as shown in AP1000 DCD Table 3.2-2, the annex building area outlined by columns E-I.1 and 2-13 is classified as seismic Category II. The annex building area outlined by columns A-D and 8-13, as well as column A-G and 13-16 is classified as nonseismic. For design purposes, only the portion identified as seismic Category II is designed following the seismic Category I structures acceptance criteria. The applicant stated that the portions of the annex building classified as nonseismic are not adjacent to the NI, and their collapse will not cause the nonseismic structure to strike a seismic Category I SSC, nor will their collapse impair the integrity of seismic Category I SSCs. The applicant further stated that the nonseismic portion of the annex building is only one story, with roof elevations below 36.7 m (120 ft). If this portion of the annex building failed, it would not cause any failure to the seismic Category II portion that could impair the integrity of the seismic Category I structures.

The staff reviewed the response and determined that additional information was needed about the seismic model used for evaluation of the seismic Category II portion of the annex building; specifically, how the nonseismic portion is incorporated in the model. During the April 2009 audit, the applicant presented pictures of the annex building, showing the seismic Category II and nonseismic portions. The applicant confirmed to the staff that failure of the nonseismic portion is not a safety concern. The applicant stated that the small, single story nonseismic section will be included in the Category I-equivalent seismic analysis of the annex building. The applicant agreed to submit a revision to its earlier response. In a letter dated August 11, 2009, the applicant submitted its revised response, providing the clarifications requested by the staff. Therefore, RAI-SRP3.7.1-SEB1-15, part (b) was resolved.

AP1000 DCD, Revision 17, Section 3.7.2.8.3, describes the design of the turbine building. The applicant revised the description of the turbine building to state that the south end of the turbine building is separated from the rest of the turbine building by a 0.61 m (2 ft) thick RC wall that provides a robust structure around the first bay. This wall isolates the first bay of the turbine building from the general area of the turbine building and from the adjacent yard area. The applicant defined the seismic classification of the turbine building as nonseismic. The staff noted an inconsistency in the turbine building description. AP1000 DCD, Revision 15, Section 3.7.2.8.3, stated "…the major structure of the turbine building is separated from the nuclear island by approximately 18 feet." However, in AP1000 DCD, Revision 17, Section 3.7.2.8.3, this statement and additional descriptive information about the turbine building were deleted. Based on the information in Revision 17, the staff could not determine whether the original classification of the turbine building as nonseismic is still valid.

In RAI-SRP3.7.1-SEB1-15, part (c), the staff requested that the applicant provide the technical basis for not classifying the turbine building as seismic Category II, considering its proximity to the NI and the infeasibility of demonstrating the acceptability of a collapse.

In its initial response dated February 6, 2009, the applicant stated that during the HR certification of the AP1000, the NRC reviewed the classification of the turbine building as a nonseismic structure. The NRC concluded from this review (NUREG-1793) "that the method and criteria used for the design of the turbine building will prevent, during a SSE event, the turbine building to jeopardize the safety function of the NI structure, and was therefore acceptable." This conclusion was reached after the applicant agreed to modify the analysis and design requirements to:

- Upgrade the UBC seismic design from Zone 2A, importance Factor of 1.25, to Zone 3 with an Importance Factor of 1.0 in order to provide margin against collapse during the SSE.

- To use eccentrically braced steel frame structures meeting the requirements given in AP1000 DCD Section 3.7.2.8.3.

The applicant further stated that the turbine building is designed as an eccentrically braced frame structure under the guidance of the UBC and is, by the principle of the code, therefore, designed to deform during the design seismic event rather than collapse. The methods and criteria that were agreed to with the NRC have not changed and are given in AP1000 DCD Section 3.7.2.8.3, Revision 17.

The staff reviewed the response and determined that the applicant had not addressed the significance of the change in the description of the turbine building from Revision 15 to

Revision 17. During the April 2009 audit, the applicant presented pictures of the turbine building, showing: (1) the recent addition of a new seismic Category II portion, which is in close proximity to the NI; and (2) the existing nonseismic portion, which is at a sufficient distance from the NI that failure is not a safety concern.

The applicant stated that any effects of the nonseismic sections of the turbine building on the Category II section of the turbine building will be included in the Category I-equivalent seismic analysis. The applicant agreed to submit a revision to its earlier response. In a letter dated August 11, 2009, the applicant submitted its revised response, providing the clarifications requested by the staff. Therefore, RAI-SRP3.7.1-SEB1-15, part (c), is resolved. In a subsequent revision to the AP1000 DCD, the applicant identified the new seismic Category II portion of the turbine building, which resolves this issue.

During the April 2009 audit, the staff and the applicant also discussed a related issue, concerning the effects of structure-soil-structure interaction (SSSI)I between the NI and the adjacent Category II structures. These adjacent Category II structures could rest on compacted backfill, with Vs significantly below 1000 fps. The applicant formally submitted its approach in a revised response to RAI-SRP3.7.1-SEB1-15, dated August 11, 2009, which included a discussion of how 2D analysis results will be scaled to simulate 3D behavior in the SSSI response. The staff reviewed the applicant's approach for performing SSSI analyses of buildings adjacent to the NI, and finds the approach acceptable. However, no analysis results were included in the RAI response. This was identified as Open Item OI-SRP3.7.1-SEB1-15 in the SER with open items.

In a follow-up response submitted July 28, 2010, the applicant provided results of the assessment of SSSI for buildings adjacent to the AP1000 NI. The seismic analyses were performed primarily using 2D [ ] models, as shown in Figures RAI-SRP3.7.1- SEB1-15-3 and RAI-SRP3.7.1-SEB1-15-4, included in the response, but the results were corrected by using a 3D-2D effect factor, which was developed using 3D [ ] models of the buildings on rigid foundations, as shown in Figure RAI-SRP3.7.1-SEB1-15-5, included in the response. Three soil cases were analyzed: UBSM, SM, and SS.

The applicant stated that the seismic Category II buildings are designed using the envelope of foundation input response spectra (FIRS) from the AP1000 design basis HR and soil cases, as well as the AP1000 HRHF spectra. The HRHF plant-grade spectra are generated using backfill soil profiles corresponding to Vs of 152.4 m/s (500 fps), 213.36 m/s (700 fps), and 304.8 m/s (1000 fps) at plant grade. The backfill Vs profiles extend from basemat El. 18.4 m (60.5 ft) to grade El. 30.5 m (100 ft). The applicant made a comparison of the resulting forces (axial and shear) and moments and showed, in Figures RAI-SRP3.7.1-SEB1-15-13 and RAI-SRP3.7.1-SEB1-15-14 of the response, that the forces and moments are controlled by the CSDRS demand rather than the HRHF demand. Also in the July 28, 2010, letter response, the applicant proposed to revise AP1000 DCD Section 3.7.2.8.4 to provide screening criteria for the COL applicant for determining whether site-specific analysis is required. If the criteria below are not met, then the COL applicant can perform site-specific analyses to demonstrate that its site-specific seismic Category II foundation seismic response spectra are less than the AP1000 annex building and turbine building first bay generic design envelope foundation spectra. The screening criteria are:

1. The site meets Section 2.5.4.5 AP1000 DCD soil uniformity requirements.

2. For soil sites, the site GMRS is enveloped by the AP1000 CSDRS with soil profiles SS, SM, UBSM, SR, FR, and HR.

3. For HRHF sites, the site GMRS is enveloped by the AP1000 HRHF response spectra with a minimum backfill surface Vs of 500 fps, and a minimum lateral extent of the backfill corresponding to a line extending down from the surface at a one horizontal to one vertical (1H:1V) slope from the outside footprint limit of the seismic Category II structure.

4. The bearing capacity with appropriate factor of safety is greater than or equal to the bearing demand.

Based on the applicant's SSSI analysis results, and the applicant's criteria for requiring site-specific analysis, the staff finds that the applicant's approach to developing seismic demands on seismic Category II structures is acceptable. Consequently, RAI-SRP3.7.1-SEB1-15 and the associated open item are resolved. In a subsequent revision to AP1000 DCD Section 3.7.2.8.4, the applicant described the screening criteria for site-specific analysis, which resolves this issue.

3.7.2.9 Conclusion

The staff concludes that Revision 19 to the AP1000 DCD continues to support the seismic system analysis for Category I SSCs to meet the applicable NRC regulations for the AP1000 DC.

The revision to the AP1000 DCD provides sufficient information to satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A, for the seismic design and analysis aspects for Category I SSCs to be used in the AP1000 reactor.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each COL applicant would have to address these issues individually.

3.7.3 Seismic Subsystem Analysis

NUREG-0800 Section 3.7.3, "Seismic Subsystem Analysis," provides guidelines for the staff to use in reviewing issues related to seismic design/analysis of subsystems. This review focused on such subsystems as the miscellaneous steel platforms, steel frame structures, tanks, cable trays and supports, HVAC ductwork and supports, and conduit and supports. Section 3.7.3, "Seismic Subsystem Analysis" of the AP1000 DCD Revision 15, was accepted in the staff's safety evaluation for the HR site DC, as documented in NUREG-1793. The AP1000 DCD, Revisions 16 through 19, made no changes to AP1000 DCD Section 3.7.3. The staff considers that its previous safety evaluation of AP1000 DCD Section 3.7.3 remains valid.

AP1000 DCD Section 3.7.2 describes the applicant's seismic analysis methods for large atmospheric storage tanks, such as the PCCWST. The PCCWST is located on the top of the shield building and is an integral part of the shield building. The applicant described the modeling and analysis approach for the PCCWST in AP1000 DCD Appendix 3G and TR-03. The staff's review identified the need for additional information. The assessment of this issue is in Section 3.7.2.3 of this SER.

3.7.4 Seismic Instrumentation

This section of NUREG-1793 is unchanged by the AP1000 DCD amendment.

3.7.5 Combined License Action Items

In AP1000 DCD Revision 18, Section 3.7.5.2 "Post-Earthquake Procedures," the applicant added the following commitment to resolve an issue related to the new and spent fuel racks seismic response evaluation:

> An activity of the procedures will be to address measurement of the post-seismic event gaps between the new fuel rack and the walls of the new fuel storage pit and between the individual spent fuel racks and from the spent fuel racks to the spent fuel pool walls and to take appropriate corrective action if needed (such as repositioning the racks or analysis of the as-found condition).

The staff assessments are in Sections 9.1.1 (new fuel rack) and 9.1.2 (spent fuel racks) of this report.

3.7.6 Seismic Design Conclusions

The staff concludes that the proposed amendment to the AP1000 DC, related to the seismic design of Category I SSCs, as described in the evaluation above, is acceptable because it satisfies the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A.

Revision 19 to the AP1000 DCD provides sufficient information to satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1; 10 CFR Part 50, Appendix S; and 10 CFR Part 100, Appendix A for the seismic design and analysis aspects for Category I SSCs to be used in the AP1000 reactor.

The applicant proposed changes to the AP1000 DCD that provide the seismic design and supporting analysis for a range of soil conditions representative of expected applicants for a COL referencing the AP1000 design. As a result, the certified design can be used at more sites without the need for departures to provide site-specific analyses or design changes, resulting in a more uniform analysis and seismic design for all the AP1000 plants. Providing the information that demonstrates the adequacy of the seismic design for a wider range of soil conditions increases the standardization of this aspect of the design. In addition, these changes reduce the need for COL applicants to seek departures from the current AP1000 design since most sites do not conform to the currently-approved hard rock sites. Therefore, the change increases standardization and meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

3.8 Design of Category I Structures

The staff has reviewed the adequacy of the design of Category I structures of the applicant's AP1000 DCD, Revisions 16 and 17 for the standard plant using the guidance provided in Sections 3.8.1, 3.8.2, 3.8.3, 3.8.4 and 3.8.5 of NUREG-0800.

The NRC issued NUREG-1793 in September 2004 for AP1000 DCD, Revision 15. Subsequent to the issuance of NUREG-1793, the applicant submitted Revisions 16 and 17 of the AP1000 DCD. Additionally, the following TRs were reviewed:

(1) TR-09, "Containment Vessel Design Adjacent to Large Penetrations," APP-GW-GLR-005

(2) TR-57, "Nuclear Island: Evaluation of Critical Sections," APP-GW-GLR-045

(3) TR-44, "New Fuel Rack Design & Structural Analysis," APP-GW-GLR-026

(4) TR-54, "Spent Fuel Storage Rack Structure/Seismic Analysis," APP-GW-GLR-033

(5) APP-1200-S3R-003, "Design for the AP1000 Enhanced Shield Building"

(6) TR-85, "Nuclear Island Basemat and Foundation," APP-GW-GLR-044

(7) TR-113, "AP1000 Containment Vessel Shell Material Specification," APP-GW-GLN-113

With these revisions, the applicant is seeking to make changes in the following areas: (1) steel containment; (2) concrete and steel internal structures of steel containment; (3) other seismic Category I structures; and (4) foundations. The specific changes in each area are evaluated by the staff using the NUREG-0800 sections identified above.

3.8.1 Concrete Containment

This section is not applicable to the AP1000 design since the AP1000 uses a steel containment.

3.8.2 Steel Containment

Using the regulatory guidance in NUREG-0800 Section 3.8.2, "Steel Containment," the staff reviewed areas relating to steel containments or to other Class MC steel portions of steel/concrete containments. The specific areas of review provided in NUREG-0800 Section 3.8.2 are as follows: (1) description of the containment; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, and special construction techniques; (7) testing and inservice surveillance program; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 Section 3.8.2 and the following SER sections provide the staff's evaluation for the relevant areas.

In its previous evaluations of AP1000 DCD, Section 3.8.2, the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 1; in GDC 16, "Containment Design"; in GDC 51, "Fracture Prevention of Containment Pressure Boundary"; and in GDC 53, "Provisions for Containment Testing and Inspection." The staff found that the AP1000 containment design was in compliance with these requirements, as referenced in NUREG-0800, Section 3.8.2, and determined that the design of the AP1000 containment, as documented in the AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria. In its previous evaluations of AP1000 DCD Section 3.8.2, the staff also concluded that satisfaction of the relevant requirements of GDC 2; GDC 4; and GDC 50, "Containment Design Basis," will be demonstrated upon completion of the ASME design report by the COL applicant.

In AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.2 of the certified design:

1. As a result of the extension of the AP1000 design from hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the Nuclear Island (NI) structures (containment, auxiliary, and shield buildings) were performed. The design of the steel containment structure for seismic loading relies upon the use of the equivalent static method, in which the acceleration profile calculated from the dynamic seismic analysis of a stick model representation of the steel containment is applied as a static load (mass times acceleration). The dynamic seismic re-analyses of the AP1000 NI, to extend the seismic design basis to soil sites, includes the same stick model representation of the steel containment. In TR-09, the applicant compared the corresponding acceleration profiles obtained from the soil-structure interaction analyses for the various soil sites to the original hard rock acceleration profile used to design the steel containment. On the basis of this comparison, the applicant concluded that the steel containment design is adequate for the range of soil sites considered.

2. The applicant eliminated the COL information item for design of the containment vessel adjacent to large penetrations. The basis for this change is documented in TR-09. The applicant indicated that the applicable changes have been incorporated into the DCD. Therefore, the combined license application (COLA) applicants are no longer required to address this item.

3. Section 3.8.2.7 of DCD Revision 16 was revised to remove the requirement that the in-service inspection of the containment vessel will be performed in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWE, and that this is the responsibility of the COL applicant. This requirement was replaced by the statement that the in-service inspection of the containment vessel will be performed.

4. The applicant undertook efforts, based on feedback from the staff transmitted in an NRC letter dated October 15, 2009, to redesign the shield building. The applicant revised the design of the shield building and submitted the details of this redesign in a separate shield building report which accounts for the revised NI model subjected to seismic and other applicable loads.

The staff has performed a confirmatory seismic analysis of the NI and discovered errors in the applicant's model used in the SSI seismic analyses. These errors occurred during the conversion of the [ ] NI20 model to the [ ] NI20 model used in the SSI analyses. The applicant indicated that it would correct the model and rerun the seismic SSI analyses. The new seismic SSI analysis was submitted on March 22, 2010, as APP-GW-S2R-010, Revision 4 (TR-03). The staff finds that both seismic loads (member forces) for structures and the design-basis ISRS have changed at some locations. The applicant's reanalysis results and RAIs, discussed in Sections 3.8.2 through 3.8.5 of the SER and the shield building SER, reflect the results of the reanalysis.

3.8.2.1 Description of the Containment

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that Figure 3.8.2-4, Sheet 6 of 6, which presents a typical containment electrical penetration, has been revised in TR-134, Revision 0. In RAI-SRP3.8.2-SEB1-06, the staff requested that the applicant explain why wedge supports on the outside of containment are used for this penetration. If they provide support to the containment penetration in the vertical and/or horizontal directions, the staff asked how the containment deformation is due to thermal and other loads accommodated or considered in the analysis. The applicant was also requested to address this item for other penetrations where this issue is applicable.

In a letter dated February 19, 2009, the applicant stated that in Figure 3.8.2-4 of the AP1000 DCD, Revision 17, the typical containment electrical penetration design was replaced with a design that does not include wedge supports at the shield building end. AP1000 DCD, Revision 17, Sections 3.8.2.1.6 and 3.8.2.4.2.5, also include revisions to information on the electrical penetrations. The staff reviewed the AP1000 DCD, Revision 17 and verified that Figure 3.8.2-4 for the typical containment electrical penetration design does not include wedge supports, and, thus, eliminates an undue constraint on the penetration. Therefore, the staff finds that RAI-SRP3.8.2-SEB1-06 is resolved.

3.8.2.2 Applicable Codes, Standards, and Specifications

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that Section 3.8.2.2, as well as other sections of the DCD related to structures; refer to AP1000 DCD Section 1.9 for discussion of conformance with RGs. The staff finds that for RG 1.7," Control of Combustible Gas Concentrations in Containment," and RG 1.57," Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," the AP1000 DCD is in accordance with earlier revisions of the RGs. The AP1000 DCD indicates that RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," is not applicable to the AP1000 DC and that Section 17.5 of the AP1000 DCD defines the responsibility for a plant maintenance program. RG 1.199, "Anchoring Components and Structural Supports in Concrete," which is identified as another applicable guide in NUREG-0800 Section 3.8, is not described at all in Section 1.9 of the AP1000 DCD.

In RAI-SRP3.8.2-SEB1-02, the staff requested that the applicant indicate whether the design, construction, and inspection of the AP1000 plant are in accordance with the current RGs and whether RG 1.199 was used to meet the NRC's regulatory guidance for the design, evaluation, and QA of anchors (steel embedments).

In a letter dated April 17, 2009, the applicant provided its response to this RAI. The staff's assessment of the response for each RG is discussed below:

<u>RG 1.7</u>

The applicant's response indicated that the current AP1000 certified design is consistent with Revision 3 of RG 1.7 (issued in March 2007). The AP1000 containment design is a passive system, using convective mixing. Design features promote free circulation of the containment atmosphere. NUREG-1793 documents an analysis of the effectiveness of the passive mixing.

The staff found that the applicant did not discuss whether the hydrogen generated loads were evaluated in accordance with RG 1.7 for the containment acceptance criteria and RG 1.57 for the applicable load combinations.

<u>RG 1.57</u>

The applicant's response indicated that RG 1.57, Revision 1 (issued in March 2007) endorses ASME Boiler and Pressure Vessel Code (B&PV), Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NE, "Class MC Components," 2001 Edition with 2003 Addenda and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with 2003 Addenda.

The applicant's response also indicated that the CV is designed to meet the requirements of ASME B&PV Code, Section III, 2001 Edition including the 2002 Addenda. The 2003 Addenda did not include any requirements that impact the design of the CV described in the AP1000 DCD. There are only two changes (which are in Subsection NE-5000, "Examination") and they are related to the examination of the welds and do not impact the design. Therefore, the applicant concluded that the CV design is in conformance with this RG.

Since the response did not discuss the regulatory positions in RG 1.57, the applicant was requested to specifically confirm whether all of the regulatory positions presented in RG 1.57, Revision 1, have been satisfied for the AP1000 plant.

<u>RG 1.199</u>

The response indicated that RG 1.199, Revision 0, was issued in November 2003, to provide guidance to licensees and applicants on methods acceptable to the staff for complying with the NRC's regulations in the design, evaluation, and QA of anchors (steel embedments) used for component and structural supports on concrete structures. As a result of studies and tests performed, questions were raised regarding the design methodology used in Appendix B to American Concrete Institute (ACI)-349-80, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 1980. After an extensive review of available test data, the ACI 349 Code committee issued a revision to ACI 349, Appendix B in February 2001.

RG 1.199 generally endorses Appendix B to ACI 349-01, with exceptions in the area of load combinations.

– The AP1000 NI concrete structures are designed to meet the requirements of the ACI 349-01 Code, including Appendix B on the design of anchors in concrete.

– Following the release of this RG, the load combinations used in the design of NI concrete structures were reviewed and approved by the NRC in the AP1000 DC for the HR sites.

The attached table to the RAI response provided itemized conformance with the regulatory positions of this RG.

In the RAI response above, the applicant did not provide any information on the provisions in RG 1.160 (10 CFR 50.65, "Maintenance Rule").

In the audit conducted during the week of May 4, 2009, the staff discussed with the applicant all the missing information associated with the above key RGs. In a letter dated September 29, 2009, the applicant transmitted a revised RAI response, which provided additional information. The staff reviewed the response and determined that it did not fully address all of the concerns related to the RGs. Therefore, the applicant was requested to address the following remaining items:

1. Explain whether the regulatory positions in RG 1.7, Revision 3 and RG 1.57, Revision 1, related to containment structural integrity under the hydrogen generated pressure loads, were satisfied or provide justification for the use of alternate methods.

2. Explain whether the regulatory positions in RG 1.57, Revision 1, related to the design limits and load combinations, were met.

3. Document in the AP1000 DCD the testing and inservice surveillance programs for plant structures. Monitoring and maintenance criteria are identified in NUREG-0800 Sections 3.8.1 through 3.8.5. With the exception of containments, each of these sections identifies that RG 1.160 is applicable. Therefore, confirm that RG 1.160 is applicable for the maintenance of structures at the plant and confirm that it will be followed when implementing 10 CFR 50.65. Also, revise the AP1000 DCD to reflect the applicability of RG 1.160, Revision 2. The performance of inservice inspection of containment is required by 10 CFR 50.55a, “Codes and standards,” and ASME B&PV Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.

4. Revise the AP1000 DCD to indicate that RG 1.199 (2003) is applicable for anchoring components and structural supports in concrete for the AP1000 plant.

In response to the above requests, the applicant’s letters dated July 2, and August 25, 2010, indicate that the AP1000 CV design is consistent with the guidance of RG 1.7, Revision 3, and RG 1.57, Revision 1. Details of the methods used to address the hydrogen generated loads, load combinations, and design limits for containment design are presented in the response to RAI-SRP3.8.2-SEB1-03. Since the design of the CV is consistent with these two RGs, the staff finds that Items 1 and 2 identified above have been adequately addressed.

To address the inservice inspection of plant structures, the applicant proposed to revise the text in AP1000 DCD Sections 3.8.3, 3.8.4, 3.8.5 and 3.8.6, and in AP1000 DCD Tables 1.8-2 and 1.9-1, to indicate that the COL applicant is responsible for establishing a structures inspection program consistent with the maintenance rule in 10 CFR 50.65 and the guidance provided in RG 1.160. This addresses the inservice testing, inspection, or special maintenance requirements for the seismic Category I and seismic Category II structures. Since the AP1000 DCD will be revised to identify the requirements for the COL applicants to develop the inservice inspection and maintenance program for structures, the staff concludes that Item 3 has been adequately addressed. The staff’s evaluation of the inservice inspection requirements for containment is discussed later in Section 3.8.2.6 of this SER.

To address Item 4, the applicant proposed to revise the text in AP1000 DCD Sections 3.8.3, 3.8.4 and 3.8.5, and in AP1000 DCD Table 1.9-1, to indicate that the design of anchorage to concrete is in accordance with ACI 349-01, Appendix B, and is in conformance with RG 1.199, Revision 0. Since the AP1000 DCD will be revised to require that concrete anchors will be designed in accordance with RG 1.199, Revision 0, the staff concludes that Item 4 has been adequately addressed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. The staff's evaluation of the inservice inspection requirements for containment is discussed later in Section 3.8.2.6 of this report.

In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves the above issues.

3.8.2.3 Loads and Load Combinations

During the review of AP1000 DCD Tier 2, Revision 16, the staff identified in RAI-SRP3.8.2-SEB1-03 a concern that Table 3.8.2-1 does not include several load combinations that are applicable to the CV design. These missing load combinations are described in 10 CFR 50.44, "Combustible gas control for nuclear power reactors"; RG 1.57; RG 1.7; and NUREG-0800 Section 3.8.2.II, Acceptance Criteria 3.B.iii. In a letter dated February 19, 2009, the applicant provided a response to this RAI. The response provided the technical basis for not considering the load combination for post flooding condition and also explained how the loading combination for external pressure due to inadvertent actuation of the fan coolers was considered. Further, the load combination with OBE for fatigue consideration was not required because the conditions specified in the ASME B&PV Code, Section III, Division 1, Subsection NE were satisfied. However, the staff determined that insufficient information was provided to explain the remaining missing load combinations and the external pressure loading imposed on the containment.

In a letter dated February 17, 2010, the applicant provided a revised response to address the remaining questions on the missing load combinations and the question on the correct external pressure to be used for the containment design. Based on the staff's review of this RAI response and the related response to RAI-TR09-08, Revision 4, the staff determined that several items still needed to be addressed. Therefore, in a follow-up RAI, the staff requested that the applicant explain why the load combinations that combine wind load with design pressure load and combine tornado wind load plus external pressure load do not appear in the proposed revision of AP1000 DCD Table 3.8.2-1. Also, the AP1000 DCD table should identify the values for the different pressures and the corresponding temperatures inside and outside containment that are used in each of these load combinations. In addition, the applicant was requested to clarify the response given regarding the hydrogen generated load evaluations for containment. These clarifications are needed to ensure that the applicable loads and load combinations described in 10 CFR 50.44; RGs 1.57 and 1.7; and NUREG-0800 Section 3.8.2, were considered.

In response to the above requests, the applicant's letters dated July 2, and August 25, 2010, indicate that the design wind load is small, within the operating pressure of the containment, which ranges from -1.38 to 6.89 kPag (-0.2 to 1.0 pounds per square inch gauge (psig)). This occurs because the shield building, which surrounds the containment, has limited openings in the vent area at the top of the cylindrical shield building wall. Therefore, the load combination that combines design wind load plus internal design pressure of 406.8 kPag (59 psig) is not included in Table 3.8.2-1. For the load combination of tornado wind load plus external pressure, the RAI response indicates that the effects of the tornado wind load for the AP1000 containment reduces the external pressure. Therefore, there is no need to consider this load combination. The staff finds that the RAI response for these two load combinations is acceptable because the effect of the wind load is considered to be negligible and the tornado load reduces the effect of the containment external pressure load.

For the definitions of the different pressures and corresponding temperatures inside and outside containment that are used in the load combinations presented in AP1000 DCD Table 3.8.2-1, the RAI response indicates that they are presented in the response to RAI-TR09-08, Revision 5. The staff confirmed that the four different pressures and temperatures are defined in the response to RAI-TR09-08. The adequacy of these pressure and temperatures is evaluated separately under the staff's assessment of RAI-TR09-08.

The RAI response provided clarifications and also proposed to make revisions in the AP1000 DCD to explain how the hydrogen generated pressure and hydrogen burn loadings were considered in accordance with 10 CFR 50.44. In addition, as noted in the staff's evaluation of RAI-SRP3.8.2-SEB1-02 above, the design of the AP1000 CV for hydrogen generated loadings is consistent with the guidance of RG 1.7, Revision 3, and RG 1.57, Revision 1. The staff finds that the information provided and the proposed changes to the AP1000 DCD are acceptable because the design is performed in accordance with 10 CFR 50.44, applicable RGs, and is consistent with NUREG-0800 Section 3.8.2. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and tables, which resolve this issue.

3.8.2.4 Design and Analysis Procedures

During the review of AP1000 DCD Tier 2, Revision 16, the staff identified that Section 3.8.2.4.1.2, which describes the local analyses for the penetrations of the steel containment, has been revised from its previous revision. The revision relies on the use of a new 3D finite element model of the entire containment, which includes the penetrations rather than using separate localized models of the penetrations. In RAI-SRP3.8.2-SEB1-04, the staff requested that the applicant provide a more detailed explanation of: (1) the new 3D finite element model of the entire containment described in Section 3.8.2.4.1.2 used for the local evaluation near penetrations; and (2) the axisymmetric model described in Section 3.8.2.4.1.1 and Appendix 3G, which is used for the analysis of the containment in regions away from penetrations. This information is needed to ensure that the revised model of the entire containment, developed for local analysis of penetrations, is adequate to capture the containment response.

In a letter dated April 29, 2009, and in a subsequent letter dated July 7, 2009, the applicant provided information to address this RAI. The staff reviewed this response and concluded that the applicant has provided a description of the 3D finite element model of the entire containment, and a description of the finite element model of the containment used for the local evaluation near large penetrations. The response indicated that more detailed information is presented in TR-09. The staff's evaluation of TR-09 is presented below. The staff reviewed the RAI response and concluded that the analysis approach is consistent with industry methods and guidance presented in NUREG-0800 Sections 3.7 and 3.8. In the July 7, 2009, RAI response, the applicant proposed several changes to be included in a future revision of the AP1000 DCD. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

Containment Penetrations Technical Report TR-09

The applicant submitted TR-09 (current version is Revision 5, March 2011) to summarize the design of CV reinforcement adjacent to large penetrations. The design of the penetrations in the TR-09 report also considers the results of the seismic evaluations conducted to extend the applicability of the AP1000 CV design to soil sites.

The applicant completed the design and analyses of the CV reinforcement for the large penetrations (two equipment hatches and two airlocks), and submitted the evaluation to the NRC as TR-09, Revision 0 in May 2006. However, the main steam and feedwater penetrations were not addressed in TR-09, Revision 0. In RAI-TR09-01, the staff requested that the applicant include the design and analysis details for the main steam and feedwater penetrations in TR-09.

In a letter dated September 5, 2007, the applicant indicated that Section 2.6 had been added to Revision 1 of TR-09, describing the design of the main steam and feedwater penetration reinforcement, and that the penetration assemblies are connected to the vessel by expansion bellows, thus preventing significant cyclic thermal and mechanical loading in the SCV.

Subsequently, during the October 2007 audit, the applicant provided report number APP-MV50-S2C-012, Revision 2, "Design of Containment Vessel Penetration Reinforcement," which included the detailed design calculations for the main steam and feedwater penetration reinforcement. The staff later reviewed this report and found that it adequately described the design of penetration reinforcement for the main steam, feedwater, and the start-up feedwater penetrations. During the October 2007 audit, the staff raised a concern that TR-09, Revision 1, did not address the fuel transfer tube penetration. The staff requested that the applicant provide information related to the design of the fuel transfer tube penetration comparable to the level of detail provided for the main steam and feedwater penetrations.

In a letter dated June 4, 2009, the applicant transmitted TR-09, Revision 3, which included the additional section on the design of containment penetration reinforcement for other penetrations, including the fuel transfer tube penetration. The staff reviewed TR-09, Revision 3 and concluded that sufficient information was provided to describe the design procedure for the other mechanical and electrical penetration reinforcements. The staff noted that the design procedure is consistent with accepted analytical methods for design of containment penetration reinforcements and is in accordance with the provisions of the ASME B&PV Code, Section III, Subsection NE, for metal containments.

On the basis that the applicant completed and documented the design of the major containment penetrations and documented the design procedure for the other containment penetrations, in accordance with the provisions of the ASME B&PV Code, Section III, Subsection NE, for metal containments, the staff considers RAI-TR09-01 resolved.

In TR-09, Revision 0, the applicant attempted to justify the use of seismic loading derived from the initial HR site condition for the design/analysis of containment penetrations for soil sites. However, the information provided was insufficient for the staff to conduct its review for the extension of the evaluation for soil sites. Therefore, in RAI-TR09-02, the staff requested that the applicant provide the necessary quantitative information in TR-09 to specifically demonstrate the design adequacy of containment penetrations for all soil conditions.

In its response dated September 5, 2007, the applicant indicated that with the exception of the large penetrations (equipment hatches and personnel airlocks), the CV design was completed for the HR site condition and was reviewed by the NRC during the HR DC, and that this design has not changed. The applicant referenced comparisons included in TR-09, Revision 1, demonstrating that the HR design forces are still applicable. The staff reviewed Figure 2-10 of TR-09, Revision 1, which compares member force and moment results from the dynamic analyses for all soil cases, to the certified HR design member forces and moments. The HR design values envelop the corresponding values for all soil sites. On this basis, the staff concluded that the overall design of the CV, based on the HR site, is also acceptable for the range of soil sites evaluated by the applicant. Therefore, RAI-TR09-02 is resolved.

Since design details for the penetrations included in TR-09, Revision 0, were not provided, the staff requested in RAI-TR09-03 that the applicant include appropriate design information (geometry, material and material properties, dimensions and wall thicknesses) for each penetration in TR-09, and specify the ASME B&PV Code, Class MC jurisdictional boundaries for each penetration.

In a letter dated September 5, 2007, the applicant indicated that typical design information for the penetrations is provided in the AP1000 DCD. This material has now been included in Appendix A of the TR-09 report. Penetration assemblies, such as those shown in the upper figure on AP1000 DCD Figure 3.8.2-4 (Sheet 4 of 6), are ASME B&PV Code Class 2. Expansion bellows and guard pipes are ASME B&PV Code Class 2 or Class MC. The penetration assemblies are welded to sleeves that are ASME B&PV Code Class MC. Process piping welded directly to the vessel, such as shown in the lower figure in AP1000 DCD Figure 3.8.2-4 (Sheet 4 of 6), is ASME B&PV Code Class 2.

The material of construction is SA738 Grade B for the vessel shell, insert plates and nozzle necks of penetrations with inside diameters greater than 60.96 cm (24 in). For penetrations less than 60.96 cm (24 in) inside diameter and greater than 5.08 cm (2 in) nominal diameter, forgings of SA350 LF2 material are used for the nozzle neck.

Other design requirements for the mechanical penetrations, as stated in the applicant's letter dated September 5, 2007, are as follows:

- Design and construction of the process piping follow the ASME B&PV Code, Section III, Subsection NC. Design and construction of the remaining portions follow the ASME B&PV Code, Section III, Subsection NE. The boundary of jurisdiction is according to the ASME B&PV Code, Section III, Subsection NE.

- Penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

- Guard pipe assemblies for high-energy piping in the containment annulus region between the containment shell and shield building that are part of the containment boundary are designed according to the rules of Class MC, Subsection NE, of the ASME B&PV Code.

- Bellows are stainless steel or nickel alloy and are designed to accommodate axial and lateral displacements between the piping and the CV. These displacements include thermal growth of the main steam and feedwater piping during plant operation, relative seismic movements, and containment accident and testing conditions. Cover plates are provided to protect the bellows from foreign objects during construction and operation. These cover plates are removable to permit inservice inspection.

The staff finds that the applicant provided design details sufficient to enable the staff to proceed with its review of the penetrations; therefore, RAI-TR09-03 is resolved.

Based on the review of TR-09, Revision 0, the staff noted that there was insufficient description of the load cases analyzed. Therefore, in RAI-TR09-05, the staff requested that the applicant describe the loads analyzed and how they were combined, and whether the containment post-loss-of-coolant accident (post-LOCA) flooding load was included in the load combinations.

In a letter dated September 5, 2007, the applicant indicated that Section 2.3 of TR-09 had been revised to describe the individual loads and their combinations; and that the post-LOCA flooding event is not considered in the load combination because it is enveloped by other design load combinations. During the October 2007 audit, the staff found that the load combinations in the AP1000 DCD and in the Containment Vessel Design Report (APP-MV50-S3R-003) are the same, but the load combinations listed in TR-09 are different. The staff requested that the applicant explain the differences or demonstrate that they are all consistent.

The adequacy of the containment load combinations is also addressed under RAI-SRP3.8.2-SEB1-03, which is evaluated in Section 3.8.2.3 of this SER. In a subsequent revision to TR-09, the applicant incorporated appropriate changes to the report text and table, which resolve the issues.

There were no results presented in TR-09, Revision 0, for buckling analyses of the containment. Therefore, in RAI-TR09-07, the staff requested that the applicant include in TR-09, Revision 0, a detailed description of buckling analysis and results.

In a letter dated September 5, 2007, the applicant indicated that Section 2.4.2.2 had been added to TR-09, Revision 1, to provide the requested information. During the May 19-23, 2008 audit, the staff reviewed calculation APP-MV50-S2C-010, Revision 0, “3D Model - Analysis of Large Penetrations,” and concluded that the buckling analyses were appropriately considered and that the calculated stresses were less than the acceptance limits. Therefore, RAI-TR09-07 is resolved.

The staff noted that AP1000 DCD, Revision 15, as well as AP1000 DCD, Revisions 16 and 17, indicate that the design external pressure is 2.9 pounds per square inch differential (psid). However, in TR-09, the applicant presented a justification for reducing the design external pressure from 2.9 psid to 0.9 psid, and stated that an estimate of the external pressure was provided in the response to DSER OI 3.8.2.1-1. Therefore, in RAI-TR09-08, the staff requested that the applicant demonstrate the design adequacy of the containment penetrations and the steel CV for a design external pressure of 2.9 psid.

In its Revision 2 response to RAI-SRP6.2.1.1-SPCV-07, dated December 14, 2009, the applicant stated that the design external pressure of 2.9 psid is used in the design load combination and the lower external pressure of 0.9 psid is a more credible external pressure used to define Service Level A and D load combinations. Because the Service Level A load combinations include thermal loads, the applicant evaluated different events at various external temperature conditions to demonstrate that 0.9 psid bounds the external pressure excursions that could occur on a cold day.

In a letter dated February 17, 2010, the applicant provided information to address questions raised regarding the temperature and external pressure loads used for design of the containment. The staff’s review of this information determined that additional information was required. In a follow-up to RAI-TR09-08, the staff requested that the applicant provide the following:

a. In Table 1 of the RAI response, the results show a trend of higher external pressure as the outside temperature increases. However, the analysis is limited to ≤ 19 degrees F, for which the external pressure is 0.98 psi. Provide the technical basis for limiting the analysis to ≤ 19 degrees F for the outside temperature.

b. After reviewing the RAI response and the proposed revision to AP1000 DCD Table 3.8.2-1, it is not clear what temperature gradient/external pressure combination is used in the Service Level A load combination notated by Footnotes 3 and 5. Describe in detail, the pressure and temperature condition used in this Service Level A load combination, and the technical basis for concluding it is the worst case. Include this information in AP1000 DCD Section 3.8.2 and in TR-09. Revise AP1000 DCD Table 3.8.2-1 footnotes to reference AP1000 DCD Section 3.8.2 that describes this loading condition.

c. The staff noted a number of inconsistencies between proposed AP1000 DCD Table 3.8.2-1 and the latest TR-09 Table 2-4, both of which identify the applicable load combinations for design of the containment structure. Revise these tables so that they are consistent, or provide the technical basis for the inconsistencies.

d. The maximum external pressure is no longer listed as 0.9 psi in the proposed revision to AP1000 DCD Table 3.8.2-1. For consistency, ensure that all references to the 0.9 psi external pressure in both the AP1000 DCD and TR-09 are appropriately revised.

Based on the applicant's letter dated July 30, 2010, much of the transient information provided previously was revised because a containment vacuum relief system was added with an actuation point of 5.5 kPa (0.8 psid). Based on the external pressure that the containment vacuum relief system can mitigate, a conservative external design pressure is defined as 11.7 kPa (1.7 psid). This design external pressure is combined with a coincident temperature of -40 °C (-40 °F) outside air temperature, which corresponds to -28 °C (-18.5 °F) for the CV shell region that is not insulated and 21.1 °C (70 °F) for the shell region that is insulated from the cold outside air. Additional information on the appropriate temperatures for this external pressure loading condition is discussed under RAI-SRP3.8.2-CIB1-01 in Section 3.8.2.5 of this SER. The applicant's July 30, 2010, letter provided the proposed changes to AP1000 DCD Section 3.8.2 related to the revised pressures and temperatures for design of the containment. The letter also indicated that TR-09 will be revised to be consistent with the AP1000 DCD changes. The staff's review of the letter concluded that the information provided in the response described the various pressure and temperature loadings to be used for design of the containment, and thus, addressed all of the staff's prior concerns for defining the pressure and temperature loads on the containment. In subsequent revisions to the AP1000 DCD and TR-09, the applicant made appropriate changes to the DCD and the report text and tables, which resolve this issue.

3.8.2.5 Materials, Quality Control, and Special Construction Techniques

In Revision 16 to the AP1000 DCD, the applicant proposed changes to the supplementary requirements of the CV shell material specification. This resulted in changes to the AP1000 DCD in Section 3.8.2.6. In a letter dated May 11, 2007, the applicant submitted TR-113, Revision 0 to provide the technical justification for the proposed changes.

Revision 15 to the AP1000 DCD, Section 3.8.2.6 specified the basic CV material as SA-738, Grade B plate. The procurement specification for this plate material is required to include supplemental requirements S17, "Vacuum Carbon-Deoxidized Steel" and S20 "Maximum

Carbon Equivalent for Weldability." The applicant has investigated the availability of SA-738, Grade B plate material (with S17 supplementary requirement) in the United States as well as in all the large, steel-producing countries in the world. The investigation determined that steel producing mills do not use an S17 process, but, rather, use a supplementary requirement S1 process to get similar high-quality, vacuum-degassed steel.

The applicant proposed to correct the AP1000 DCD in Revision 16 to specify supplementary requirement S1 instead of the currently specified supplementary requirement S17. The applicant provided the following technical justification in support of the proposed change to AP1000 DCD Section 3.8.2.6.

The use of a vacuum carbon-deoxidized (VCD) process in steel production typically applies to certain grades of chromium-molybdenum (Cr-Mo) steels where carbon contents are lower and reduced silicon content is beneficial. The VCD process allows oxygen and carbon to react in the molten steel and evolve as carbon monoxide, which is drawn off by the vacuum. While under vacuum, other gases, such as hydrogen and nitrogen, also tend to be removed from the steel. Reducing the oxygen content by VCD reduces the need for the addition of other deoxidizing additions such as silicon or aluminum. Steels treated by VCD have a specified silicon content of 0.12 percent maximum that is lower than the normally specified range of silicon content. This process is beneficial in Cr-Mo steels that are susceptible to temper embrittlement during elevated-temperature service. Silicon is one of the impurity elements that contribute to the loss of toughness. By reducing the silicon content of the steel the tendency for temper embrittlement is reduced. The use of the VCD process for vacuum degassing of SA-738 plate material was discussed with a metallurgist from a large, domestic-steel plate producer. The steel producers in the United States typically do not use VCD for plate materials like SA-738. For this reason, requiring supplementary requirement S17 to be used for the production of SA-738 plate material is somewhat of an anomaly. Therefore, the supplementary requirement S1, "Vacuum Treatment," is more appropriate for this type of material because S1 requires the steel to be made by a process, which includes vacuum degassing while molten by a suitable practice selected by the steel manufacturer or purchaser.

In addition, Revision 16 to the AP1000 DCD, Section 3.8.2.6 was changed to specify the lowest service temperature of -28 °C (-18.5 °F) instead of -26.1 °C (-15 °F), which was previously stated in Revision 15 of the AP1000 DCD. TR-113 did not specify the change to the service temperature nor provide any justification for this change in service temperature as required by 10 CFR 52.63(a)(1).

The staff reviewed the applicant's request to revise AP1000 DCD, Section 3.8.2.6 concerning the supplementary requirements of the CV shell material specification and found it acceptable because of following reasons.

The SA-738, Grade B plate material was approved for use in metal CV construction in ASME Code Case N-655, Section III, in February 2002. This plate material was also incorporated into Table 1A of Section II, Part D in the 2002 Addenda to the 2001 Edition of the ASME B&PV Code. The NRC conditionally accepted ASME Code Case N-655 in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 33 in August 2005. The conditions that the NRC placed on the use of SA-738 plate material were to specify the use of supplementary requirements S17 and S20 when using SA-738 material for CV construction. The two conditions were needed to ensure adequate material properties and weldability of the CV material. The ASME Code, Section III, exempts SA-738, Grade B, material up to 4.4 cm (1.75 in) of thickness from post-weld, stress-relief heat treatment.

Because the welds in CV material thickness up to 4.4 cm (1.75 in) thick will not be stress-relieved, higher residual stresses will be present in the welds. Also, the material will likely be procured in the quenched and tempered condition. Welding will reduce the impact properties of the material in the heat affected zone. Requiring the use of vacuum degassed steel will ensure adequate material properties because nonmetallic inclusions, such as oxides and silicates will be minimized as a result of the vacuum degassing of the steel. S17 supplementary requirement was specified to accomplish the vacuum degassing of the steel. Requiring supplementary requirement S20 and a carbon equivalent weldability check will ensure that the steel is readily weldable.

The staff specified the use of S17 for SA-738 material because at the time of the review of ASME Code Case N-655, S17 was the only requirement clearly listed in the specification that would provide for vacuum degassing of steel. Supplementary requirement S1 was also available for SA-738 plate material; however, S1 is listed in SA-20, "General Requirements for Steel Plates for Pressure Vessels," which is referenced in the SA-738 specification. Therefore, in order to impose the S1 requirement in the CV, the designer would have to specify two specifications instead of one. The purpose of the staff's condition was to specify the use of vacuum degassed steel. Imposing an S1 supplementary requirement would accomplish this goal. Furthermore, at the time of approval of ASME Code Case N-655 neither the staff nor the applicant was aware that the steel producers had limited S17 to the production of Cr-Mo steels. Since the discovery of this situation, the ASME Code has approved a revision to the ASME Code Case N-655-1, which correctly specifies the use of S1 and S20 supplementary requirements for the use of SA-738 plate material. On this basis, the staff concludes that the proposed revision to AP1000 DCD, Section 3.8.2.6 to specify supplementary requirement S1 meets the requirements of 10 CFR 50.55a and the ASME Code, Section III, and is acceptable.

In regard to the service temperature of the CV, Tier 2, Section 3.8.2.6 of the AP1000 DCD, describes the materials used to fabricate the CV. The material selected satisfies the lowest service metal temperature requirement, established by analysis for the portion of the vessel exposed to the environment when the ambient air temperature is -40 °C (-40 °F). TR-113, Revision 0, submitted by the applicant in a letter dated May 11, 2007, also revised this section to specify the lowest service temperature of -28.1 °C (-18.5 °F) instead of -26.1 °C (-15 °F), which was previously stated in Revision 15 of the AP1000 DCD. TR-113 did not specify the change to the service temperature nor provide any justification for this change in service temperature as required by 10 CFR 52.63(a)(1). In NUREG-1793, Section 3.8.2.6, the staff approved -26.1 °C (-15 °F) as the lowest service temperature based on the staff's review of the applicant's calculation APP-PCS-M3C-002, Revision 1, "AP1000 Containment Shell Minimum Service Temperature." Therefore, the staff requested that the applicant provide its reason and justification for the change in minimum service temperature of the CV in accordance with 10 CFR 52.63(a)(1), along with the analysis that supports the new service temperature proposed in Revision 16 of the AP1000 DCD. This was previously addressed in RAI-SRP3.8.2-CIB1-01.

In a letter dated July 22, 2008, the applicant stated that an additional scenario was postulated for the CV shell analysis, which determined that the CV will be subjected to a service metal temperature of -28.1 °C (-18.5 °F). This evaluation postulated that an SSE event occurred in conjunction with -40 °C (-40 °F) outside temperature and inadvertent actuation of active containment cooling. APP-GW-GLR-005 (TR-09) only described the analysis, and inadvertently did not include the corresponding service metal temperature.

Since TR-09 did not include the analysis or the service metal temperature, the staff could not confirm that -28.1 °C (-18.5 °F) was the lowest service metal temperature of the CV shell, which is fabricated from SA-738 Grade B material. This material must meet the requirements of NE-2000 for fracture toughness (Charpy V-notch test) in the as-welded condition for thicknesses up to and including 4.4 cm (1.75 in), and in the post-weld heat treated condition for thicknesses greater than 4.4 cm (1.75 in). The minimum service temperature is used to determine the testing temperature for the Charpy V-notch tests required by the ASME Code, Section III, Subsections NE 2300 and NE-4300. Previously, the applicant stated in its letter dated April 22, 2003, that the SA-738, Grade B plate material will be procured using the service metal temperature of -26.1 °C (-15 °F) (i.e., -48.3 °C (-55 °F) Charpy V-notch test temperature as required by the ASME Code, Section III, Subsections NE-4335.2(b)(2) and Tables NE-4622.7(b)-1, note (2)(b)(1)) in order to account for degradation during welding of the heat affected zone in the base material. In addition, the applicant stated in a letter dated March 13, 2003, that the previous analysis added a -13.3 °C (8 °F) conservative factor to obtain a minimum service metal temperature of -26.1 °C (-15 °F).

Therefore, the staff required additional information to verify the minimum service metal temperature including the details of the analysis (e.g., calculation methodology, assumptions made, similarities/differences from previous analysis, etc.) to confirm that -28.1 °C (-18.5 °F) is the lowest service metal temperature to ensure that the material will be tested to have adequate toughness for the design and environment the containment shell will experience. The staff also requested clarification of whether the conservative factors described in the applicant's letter dated March 13, 2003, were used in this analysis or provide justification for not including these conservative factors.

In a letter dated May 7, 2009, the applicant stated that the additional information was provided in APP-MV50-Z0C-020, Revision 0. However, the staff requested that the assumptions made along with the similarities/differences from the previous analysis (for Revision 15 of the AP1000 DCD) be addressed. In response to Revision 2 of RAI-SRP3.8.2-CIB1-01, the applicant provided in a letter dated September 17, 2009, the assumptions and differences between the analyses. The applicant stated that the original analysis for -26.1 °C (-15 °F) minimum service metal temperature in Revision 15 of the AP1000 DCD was performed by a hand calculation using a simple radial heat balance model, and then added an -13.3 °C (8 °F) conservatism factor. The minimum service metal temperature of -28.1 °C (-18.5 °F) was determined by a WGOTHIC computer code, using a free/forced convection model. This model calculated a higher heat transfer coefficient; thereby, resulting in a lower minimum service metal temperature (-28.1 °C (-18.5 °F) versus -26.1 °C (-15 °F)). The staff notes that WGOTHIC is currently used in other pressure and temperature determinations for operating reactors. In addition, WGOTHIC has its own inherent conservatisms within the computer code. Therefore, the staff determined that the use of WGOTHIC computer code is valid in determining the minimum service metal temperature for the steel containment.

In a letter dated February 17, 2010, the applicant performed a new WGOTHIC analysis documented in APP-MV50-Z0C-039, Revision 0, which used an outside temperature at -40 °C (-40 °F) and -34.4 °C (-30 °F). However, the staff notes that this analysis was not a bounding case, since it used different assumptions for the wind speeds at these two temperatures based on Duluth, Minnesota, meteorological data. The Duluth data documented the wind speed at -34.4 °C (-30 °F) to be faster than at -40 °C (-40 °F). Using these temperatures and wind speeds, the -34.4 °C (-30 °F) case resulted in a higher velocity through the annulus between the containment and air baffle, and thereby, a greater heat transfer coefficient. Therefore, based on the Duluth, Minnesota, weather records, the applicant's analysis determined that the -34.4 °C

(-30 °F) outside temperature condition resulted in minimum service metal temperature of -8.1 °C (-0.61 °F) versus a minimum service metal temperature of -13.8 °C (7.18 °F) for an outside temperature of -40 °C (-40 °F). Since the analysis in APP-MV50-Z0C-039, Revision 0 was not a bounding case, the staff requested that a bounding analysis be performed using an outside temperature of -40 °C (-40 °F) and a maximum wind speed of 77 km/h (48 mph), used in previous calculations, or provide justification for the validity of the Duluth temperature/wind speed data along with a sensitivity study.

In a letter dated May 10, 2010, the applicant provided an analysis for the loss of alternating current (ac) power (LOAC) transient using an outside temperature of -40 °C (-40 °F) with a corresponding wind speed of 48 mph, which produced a minimum service metal temperature of -27.2 °C (-16.91 °F), which is bounded by the -28.1 °C (-18.5 °F) minimum service metal temperature in the AP1000 DCD. The staff notes that the -8.4 °C (16.91 °F) temperature included a factor to compensate for any temperature uncertainty in the calculation near the air baffle plate. The bounding case used the LOAC transient in Case 11 of APP-MV50-ZOC-039, Revision 0, by adjusting the wind speed to 77 km/h (48 mph). Based on the June 18, 2010, letter, the applicant stated that the LOAC transient was the limiting event since the inadvertent activation of the containment fan cooler event is no longer credible because the fan coolers are operational. Therefore, the staff considers this to be a bounding condition in determining the minimum service metal temperature and that the -28.1 °C (-18.5 °F) temperature in the AP1000 DCD is supported by an appropriate analysis. The staff notes that in the letter dated May 10, 2010, the applicant provided a bounding calculation in lieu of justifying the current data in APP-MV50-ZOC-039, Revision 0. However, the applicant did not revise APP-MV50-ZOC-039, Revision 0, to reflect this bounding calculation, and assumes that the results depicted in APP-MV50-ZOC-039, Revision 0, are the result of record for the AP1000 DCD. The staff requests that the applicant revise APP-MV50-ZOC-039, Revision 0, to reference this bounding calculation, since the bounding case was provided in lieu of justifying the current data in APP-MV50-ZOC-039, Revision 0. The staff identifies this as Open Item OI-SRP3.8.2-CIB1-01.

In a letter dated July 9, 2010, the applicant stated that the bounding case provided in the letter dated May 10, 2010, would be incorporated into APP-MV50-ZOC-039. In addition, the applicant stated in letters dated July 30, 2010, and August 16, 2010, that the addition of a vacuum relief system does not invalidate APP-MV50-ZOC-039 for the determination of the minimum service metal temperature. The staff agrees that the bounding calculation for the minimum service metal temperature in APP-MV50-ZOC-039, as modified by letter dated July 9, 2010, is still applicable, since it is calculates the lowest possible service metal temperature corresponding with an outside temperature of -40 °C (-40 °F). This resolves Open Item OI-SRP3.8.2-CIB1-01.

However, the staff notes that Revision 17 inadvertently revised Section 3.8.2.6 of the AP1000 DCD to specify a minimum service metal temperature of -26.1 °C (-15 °F). In a letter dated June 18, 2010, the applicant proposed to change the minimum service metal temperature back to -28.1 °C (-18.5 °F), which is supported by the bounding analysis. Therefore, the staff finds this proposed change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

3.8.2.6 Testing and Inservice Inspection Requirements

During the review of AP1000 DCD Tier 2, Revision 16, the staff identified that Section 3.8.2.7 had been revised to remove the requirement that the inservice inspection of the CV would be performed in accordance with the ASME Code, Section XI, Subsection IWE, and that this is the responsibility of the COL applicant. In accordance with the guidance presented in NUREG-0800 Section 3.8.2, this information should be provided by the applicant for review by the staff. Therefore, the staff requested, in RAI-SRP3.8.2-SEB1-05, that the applicant include in the AP1000 DCD information that describes how the AP1000 containment complies with the 10 CFR 50.55a requirements and the ASME Code, Section XI for the preservice and inservice examination of the containment.

In a letter dated February 27, 2009, the applicant indicated that Section 3.8.2.7 of the AP1000 DCD would be revised to reference Section 6.6, which identifies that the COL applicant will perform inservice inspection of the containment according to the ASME Code, Section XI. Section 6.6.9.1 includes a COL information item for the COL applicant to prepare preservice and inservice inspection programs for the ASME Code systems and components.

Section 6.6 was revised in the AP1000 DCD, Revision 17 to specifically include ASME Code Class MC components. The applicant indicated that Sections 6.6.9.1 and 6.6.9.2 will be revised to also specifically include Class MC systems and components.

The staff concludes that the RAI response is acceptable because: (1) the applicant will revise AP1000 DCD Section 3.8.2.7 to reference Section 6.6, which indicates that inspection of the containment is performed in accordance with the ASME Code, Section XI and 10 CFR 50.55a; (2) AP1000 DCD Section 6.6 indicates that COL applicants will prepare the inspection program for the containment; and (3) the applicant will revise AP1000 DCD Sections 6.6.9.1 and 6.6.9.2 to require the preparation of an inspection program for Class MC (containment) systems and components. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

3.8.2.7 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the applicant's application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 containment as documented in AP1000 DCD, Revision 19, against the relevant acceptance criteria as listed above and in NUREG-0800 Section 3.8.2.

The staff concludes that the AP1000 DCD Section 3.8.2.5 revisions proposed by the applicant meet the requirements of 10 CFR 50.55a and the ASME Code, Section III, applicable RGs, and NUREG-0800 Section 3.8.2 and, therefore, are acceptable.

The staff concludes that design of the containment continues to meet all applicable acceptance criteria. In summary, based on the above discussions, the staff finds that the design of the AP1000 containment is acceptable.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each applicant would have to address these issues individually.

3.8.3 Concrete and Steel Containment Internal Structures

Using the regulatory guidance in NUREG-0800 Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," the staff reviewed: (1) description of the internal structures; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, and special construction techniques; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 Section 3.8.3 and the following SER sections provide the staff's evaluation for the relevant areas.

In its previous evaluations of AP1000 DCD, Section 3.8.3, the staff identified acceptance criteria based on the design meeting the relevant requirements in 10 CFR 50.55a; 10 CFR Part 50, Appendix A, GDC 1, GDC 2, GDC 4, and GDC 50. The staff found that the design of the AP1000 CISs was in compliance with these requirements, as referenced in NUREG-0800 Section 3.8.3 and determined that the design of the AP1000 CISs, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

In AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.3 of the certified design:

1. As a result of the extension of the AP1000 design from just hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the NI structures were performed. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the re-analyses included the additional use of response spectrum and time history methods of analysis. In DCD Revision 16, Table 3.8.3-2 was revised to include the use of the response spectrum analysis for the seismic analysis of the containment internal structures. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

2. In DCD Revision 16, the applicant removed Section 3.8.3.4.1.2 - Stiffness Assumptions for Global Seismic Analyses in the previous certified DCD. This section discussed the stiffness properties used in the seismic analyses of the containment internal structures and the auxiliary building modules. Reference was made to DCD Table 3.8.3-1, which contained the various stiffness cases for the concrete filled steel modules used for structures inside containment and the auxiliary building. This deletion of the prior text in Section 3.8.3.4.1.2 shifted the text in the sections that followed Section 3.8.3.4.1.2 (i.e., prior Section 3.8.3.4.1.3 became Section 3.8.3.4.1.2 and prior Section 3.8.3.4.1.4 became 3.8.3.4.1.3).

3. In DCD Revision 16, the applicant revised Section 3.8.3.5.7 - Design Summary Report. DCD Revision 15 indicated that the results of the evaluation would be documented in an as-built report by the Combined License applicant. In DCD Revision 16, this was revised to state that, "The results of the evaluation will be documented in an as-built summary report." Thus the phrase "by the Combined License applicant" was removed.

4. In DCD Revision 16, the applicant revised Section 3.8.3.5.8 - Design Summary of Critical Sections, in several subsections which describe the design of different

specific critical sections. This set of revisions included changes in the text portion, revisions in a number of the DCD tables, and removal of some Tier 2* information. Some of these revisions referred to Appendix H of the DCD, which is discussed below in item 5.

5. Based on the changes discussed above for DCD Section 3.8.3.5.8, the referenced DCD Appendix 3H - Auxiliary and Shield Building Critical Sections, in both DCD Revisions 16 and 17, had substantial revisions in the text, tables, and figures.

6. In DCD Revisions 16 and 17, the applicant revised Section 3.8.3.6 - Materials, Quality Control, and Special Construction Techniques. The revisions relate to the change in material for the structural modules from Nitronic 33 to Duplex 2101, and relate to the change in the industry standard from NQA-2 to NQA-1 for packaging, shipping, receiving, storage and handling of the structural modules in accordance with industry specification AISC N690.

7. In DCD Revision 17, the applicant revised Section 3.8.3.6.3 - Concrete Placement, regarding how concrete will be placed in the CA01 module inside the containment. The previous phrase in DCD Revision 15, which stated that the concrete is placed in each wall continuously from the bottom to the top was removed, and the description of the concrete placement was revised to state that concrete will be placed either through multiple delivery trunks located along the top of the wall or through windows in the module walls or pumping ports built into the module wall.

8. A new 59.5 m^3 (2100 ft^3) pressurizer is used. It has a smaller length from the outside surface of the lower head to the outside surface of the upper head. This change was made to reduce the seismic response of the pressurizer compartment.

3.8.3.1 Applicable Codes, Standards, and Specifications

During the review of AP1000 DCD Tier 2, Revision 16, the staff noted that Sections 3.8.3.2 and 3.8.4.2 describe the codes, standards, and specifications used for structural components of the AP1000. In view of the extension of the AP1000 design to soil sites, reanalysis for updated seismic spectra, design changes made to structures, and to ensure that the AP1000 meets the safety requirements in current regulatory positions, the staff, in RAI-SRP3.8.3-SEB1-01, requested that the applicant identify whether the AP1000 plant meets industry standard American National Standards Institute/American Institute of Steel Construction (ANSI/AISC)-N690-1994, Supplement 2 (2004) and the more recent versions of the applicable American Welding Society (AWS) standards than are currently listed in AP1000 DCD, Revision 16. These references are cited in the current NUREG-0800, Section 3.8, which was issued subsequent to the license application for the AP1000 DCD, Revision 16.

In the applicant's letters dated April 3, 2009, and October 22, 2009, the applicant stated that the references to AISC-N690-1994 and the other applicable codes, standards and specifications in AP1000 DCD Sections 3.8.3.2 and 3.8.4.2 have not changed from AP1000 DCD, Revision 15 to Revision 17. The applicant indicated that the staff previously accepted the technical basis for concluding that the standards listed in AP1000 DCD Section 3.8, Revision 15 provide sufficient conservatism or equivalent levels of safety. Therefore, the applicant does not intend to evaluate conformance to later editions and revisions of these codes and standards.

Since the staff previously accepted the use of the ANSI/AISC-N690-1994 and AWS standards in the certified design as described in AP1000 DCD, Revision 15 and these standards were considered to be acceptable, subject to certain supplementary requirements as stated in AP1000 DCD Section 3.8, the staff finds that these standards are also acceptable for use in the current design of the AP1000. Therefore, RAI-SRP3.8.3-SEB1-01 is resolved.

3.8.3.2 Analysis Procedures

During the review of the AP1000 DCD Tier 2, Revision 16, the staff noted that the entire Section 3.8.3.4.1.2, "Stiffness Assumptions for Global Seismic Analyses," of the AP1000 DCD, Revision 15 had been deleted. Therefore, in RAI-SRP3.8.3-SEB1-03, the staff requested that the applicant provide a description of the CIS model, the stiffness assumptions used, and the basis for the selection of the stiffness for the CIS and auxiliary building modules.

In a letter dated February 24, 2009, the applicant provided a response, which explained that the description for the model development and analysis for the CIS are provided in AP1000 DCD Section 3.7 and TR-03. As a result of the staff's review of the RAI response, several questions were identified and these items were discussed with the applicant in a conference call on May 12, 2009. The applicant was requested to clarify the information presented in the first three rows of AP1000 DCD Table 3.8-2, regarding the specific models used. In addition, the staff requested that the applicant explain whether the models were local or global and where these analyses were described in the AP1000 DCD, and the basis for selecting the module concrete stiffness values used. During the conference call, the applicant indicated that it would provide a revised RAI response to address these items.

In a letter dated October 19, 2009, the applicant provided some information regarding the stiffness values used; however, the staff determined that further justification was needed regarding the proper stiffness utilization for the modules of the CIS and for the other RC structures. The RAI response indicates that the NI model of concrete structures is based on the gross concrete section stiffness reduced by a factor of 0.8 for the consideration of the effect of concrete cracking as recommended in Table 6-5 of FEMA 356. The staff finds that Table 6-5 of FEMA 356 indicates that the factor of 0.8 is only applicable to flexural rigidity for concrete walls that are uncracked when inspected. For walls that are cracked, the stiffness reduction factor for flexure is 0.5. For shear rigidity, the FEMA table indicates that the stiffness reduction factor is 0.4 for walls that are uncracked and cracked. Therefore, it is not appropriate to reference the FEMA standard as justification for the use of the 0.8 factor. In a follow-up RAI, the applicant was requested to justify the stiffness reduction factor used in the analysis and design of RC structures and the concrete-filled steel members used for the CIS and other structures.

To demonstrate the adequacy of using the 0.8 stiffness reduction factor for the RC and concrete-filled steel members in the seismic analysis of the NI structures, the applicant performed a study. In a letter dated July 30, 2010, the applicant updated its responses to RAI-SRP3.7.1-SEB1-19 and RAI-SRP3.8.3-SEB1-03, and provided comparisons of the [ ] linear and [ ] nonlinear analysis results. The [ ] linear analysis used the [ ] stiffness reduction factor and the [ ] nonlinear analysis used a concrete cracking model, which reflected the concrete stiffness based on the degree of cracking in the finite elements. Both analyses were time-history analyses based on the envelope of the soil and rock profiles. Comparisons were made at the shield building roof elevation, shield building West wall (at grade elevation) and at four other locations in the auxiliary building. The response spectra at these six locations showed a comparison close enough to allow for a conclusion that the [ ] stiffness reduction factor is acceptable.

However, the applicant did not provide [ ] comparisons for the same locations. Since [ ] is the AP1000 design basis code, the staff believes that the [ ] to [ ] comparisons are required to validate model similarity. In an updated response to RAI-SRP3.8.3-SEB1-03, dated September 3, 2010, the applicant provided the requested comparisons between the [ ] and [ ] linear analysis results. This comparison demonstrated similarity between the [ ] and [ ] models. The applicant also provided additional information on the [ ] RC to SC connection modeling approach. This information showed that the response of this [ ] RC/SC connection compared closely with the detailed FEM representation of the RC to SC connection, which included the tie bars, reinforcement, steel plates, and concrete. The RAI response also provided markups to DCD Section 3.8.3 to incorporate the concrete stiffness reduction factor used for the CIS.

On the basis of the results of the studies discussed above, the staff concluded that the approach for addressing concrete cracking is acceptable. The applicant's study using [ ], supported by the correlation of linear results between [ ] and [ ], indicate that a reduced concrete modulus of [ ] is justified for the design-basis analysis of the concrete filled steel modules and RC sections and, therefore, is acceptable. The staff further concluded that the RC/SC connection simulation in the [ ] nonlinear analysis model provides a reasonable representation of the effect of the connection on the overall seismic response and its use is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and table, which resolve this issue.

3.8.3.3 Design Procedures and Acceptance Criteria

The staff requested, in RAI-SRP3.8.3-SEB1-04, that the applicant address concerns with the design details of the structural module connections to the RC basemat. Section 3.8.3.5.3 of the AP1000 DCD indicates that the steel plate modules are anchored to the RC basemat by mechanical connections welded to the steel plate or by lap splices. Typical details of these two options are shown on AP1000 DCD Figure 3.8.3-8, Sheets 1 and 2.

In a letter dated February 27, 2009, the applicant provided clarification of the details of the structural module connection to the basemat concrete. Correction of the connection detail on the left side of Figure 3.8.3-8, Sheet 2, and a new alternate connection detail will be included in the next update to the AP1000 DCD. Regarding the connection detail on the right hand side of Figure 3.8.3-8, Sheet 2, the staff's understanding is that this type of connection detail is not addressed by ACI 349 Code and does not provide for a direct transfer of load from the concrete to the steel module plates as do the other two alternates. Therefore, the applicant was requested to explain why the connection detail on the right side of Figure 3.8.3-8 was not removed or to provide a technical basis to demonstrate its structural adequacy. The information provided in the RAI response simply made reference to recommendations and test data given in a paper presented in a conference. In a conference call on May 12, 2009, the staff discussed the above items with the applicant, and the applicant agreed to provide a revised RAI response to address the staff's concerns.

In a letter dated March 12, 2010, a partial response was provided; however, the information still did not demonstrate the adequacy of the connection of the structural modules to the base concrete. Therefore, in a follow-up RAI, the staff indicated that, since the type of connection shown in the right side of AP1000 DCD Figure 3.8.3-8, Sheet 2, is not covered by ACI 349, the applicant should describe how the loads from the module could be properly transferred from the module to the embedded bars in the base concrete and explain how the design is performed. Also, the applicant was requested to explain why the design of the connection does not rely on the other existing option of transferring loads directly from the faceplates to the base concrete using vertical bars and mechanical connectors.

In response to the above requests, the applicant's letters dated July 30, 2010 and August 25, 2010, deleted the connection detail that does not have a direct load transfer path from the structural modules to the base concrete. In addition, a representative connection detail relying only on a direct load transfer path was proposed to be shown in AP1000 DCD Figure 3.8.3-8, Sheet 2, and all other connection alternatives would be deleted from the figure. Because the connection detail provided is identified as representative and the final design may differ to account for items such as accessibility for inspection or ease of fabrication and construction, the applicant proposed to include another note, which states that any changes to the mechanical connection detail shall maintain a direct load path to transfer loads from both sides of the module surface plates to the vertical dowel bars in the base concrete through the use of intervening plates, mechanical connectors and welds. The staff found the RAI responses are acceptable because the representative design details proposed will provide a direct load path to transfer loads from both sides of the module surface plates to the vertical dowel bars in the base concrete. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and figure, which resolve this issue.

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that AP1000 DCD, Revision 16, Tables 3.8.3-3 through 3.8.3-7 had been revised removing their identification as Tier 2*. The revised tables removed information that provided the required plate thicknesses and stress results that permit comparison to the plate thicknesses provided and allowable stress limits. In RAI-SRP3.8.3-SEB1-07, the staff requested that the applicant provide the information in the AP1000 DCD, Revision 16, for these tables equivalent to that provided in Revision 15. Also, AP1000 DCD, Revision 16, Table 3.8.3-7 replaced specific AISC interaction ratio values in Revision 15 with a notation that it is now less than 1.0 at all entries of the table. Therefore, the staff requested that the applicant present the actual interaction ratios as was done in the prior version of the AP1000 DCD.

In a letter dated March 2, 2009, the applicant provided an explanation as to why the Tier 2* information was revised in Revision 16 of the AP1000 DCD. One explanation was that these changes were communicated to the NRC in APP-GW-GLR-045 (TR-57), Revision 1, dated November 21, 2007, Chapter 5.0, "DCD Mark Up" (November 2007), and these changes were also discussed in an audit meeting in Pittsburgh. The RAI response did not provide the requested stress results and the AISC interaction ratio values. The staff reviewed the RAI response and concluded that it did not justify the elimination of the Tier 2* designation of the design information for the critical sections. The AP1000 DCD must provide a complete design for the AP1000 plant and some of this information may be identified as Tier 2* information. In a conference call on May 12, 2009, the staff discussed these issues with the applicant, which agreed to provide a revised RAI response to address the staff's concern.

In a letter dated March 15, 2010, the applicant indicated that all of the information in Table 3.8.3-7 comparable to the data presented in the same table in the AP1000 DCD, Revision 15, would be provided in the proposed mark-ups to the AP1000 DCD amendment application. The changes to the other AP1000 DCD tables were provided in the response to RAI-SRP3.8.3-SEB1-05. The staff's review of the mark-ups for Table 3.8.3-7 concluded that the information provided is comparable to the table in the AP1000 DCD, Revision 15, and that the tabulated results for the steel wall of the IRWST show the interaction ratios are all less than 1.0 in accordance with the AISC and the ASME Code stress limits. The staff met with the applicant on October 14, 2010, to discuss the applicant's proposed identification of Tier 2* items in the proposed DCD. As a result, the applicant stated it is revising the DCD to include revised Tier 2* items in Revision 2 to the response to RAI-SRP3.8.3-SEB1-07, dated October 21, 2010. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and tables, which resolve this issue.

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified several items, described in AP1000 DCD Section 3.8.3.5.8, related to the design summary of critical sections for the CIS to be addressed. These items affect Section 3.8.3.5.8.1, "Structural Wall Modules"; Section 3.8.3.5.8.2, "IRWST Steel Wall"; and Section 3.8.3.5.8.3, "Column Supporting Operating Floor." In RAI-SRP3.8.3-SEB1-05, the staff requested that the applicant explain: (1) why certain Tier 2* information and criteria were removed from the AP1000 DCD; (2) why references for CIS are made to Appendix 3H, which is applicable to auxiliary and shield building critical sections; and (3) whether the existing results in Sections 3.8.1 through 3.8.5, and associated appendices reflect the latest set of updated analyses for the revised seismic loads and other loadings.

In a letter dated March 15, 2010, the applicant addressed most of the concerns identified in this RAI. The staff's review of the response noted that most of the Tier 2* information including descriptions, criteria, member forces, required plate thicknesses and stress results, removed from Section 3.8.3.5.8 of the AP1000 DCD, Revision 17, would be restored in AP1000 DCD Sections 3.8.3.5.8.1 to 3.8.3.5.8.3 and Tables 3.8.3-4 through 3.8.3-6. Therefore, in a follow-up RAI, the staff requested that the applicant include the required plate thicknesses, which were provided in the same table in the certified design presented in the AP1000 DCD, Revision 15, and to correct the designation of the Tier 2* information in AP1000 DCD Section 3.8.3.5.8.1.

In response to RAI-SRP3.8.3-SEB1-05, the applicant's letters dated July 2, and August 25, 2010, provided proposed mark-ups to AP1000 DCD Section 3.8.3.5.8, and the corresponding tables, where the required plate thicknesses were added. The staff reviewed the proposed mark-ups to the AP1000 DCD and concluded that they were acceptable because corrections were made to include the required plate thicknesses and to correct the improper designation of the Tier 2* information.

In addition, the applicant-proposed mark-ups included new criteria, which are tolerances on certain values designated as Tier 2*, intended to explain when changes in the values presented in the critical section Tier 2* tables must be reported to the NRC. The two new criteria presented are as follows:

(1) if a change increases or decreases the design parameters (e.g., reinforcement provided, concrete strength, or steel section size), then the change must be reported to the NRC; and

(2) if changes in the values of the loads, moments, and forces in the critical section tables that are designated as Tier 2* result in a required reinforcement (or plate thickness for the containment internal structures) increase greater than 10 percent of the provided reinforcement (or plate thickness for the containment internal structures) then the increase must be reported to the NRC.

Tier 2* information is part of the DCD that cannot be changed by a license holder without prior approval. However, the criteria, proposed by the applicant for identifying when changes in values presented in the critical section Tier 2* tables identified some Tier 2* changes that would not have to be approved by the NRC; these proposed criteria are not in compliance with the regulatory requirements of 10 CFR Part 52, Appendix D, “Design Certification Rule for the AP1000 Design,” Section VIII.6.a. It should be noted that the proposed criteria for Tier 2* would also apply to AP1000 DCD Section 3.8.5.4.4, Table 3.8.5-3, and AP1000 DCD Appendix 3H, for which the applicant also planned to use its proposed criteria. The staff met with the applicant on October 14, 2010, to provide this feedback. As a result, by letter dated October 21, 2010, the applicant stated it would withdraw TR-57, and revise the DCD to include revised Tier 2* information in Revision 4 to the response to RAI-SRP3.8.3-SEB1-05, dated October 21, 2010. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and tables, which resolve this issue.

3.8.3.4 Materials, Quality Control, and Special Construction Techniques

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that AP1000 DCD Section 3.8.3.6 was revised regarding the use of different steel materials for CIS structural modules from the previously certified AP1000 design. In RAI-SRP3.8.3-SEB1-06, the staff requested that the applicant discuss the revision of materials: (1) from [ ] grade steel plates and shapes for the modules to allow the use of other grade carbon steel plates and shapes; and (2) from [ ], stainless steel plates for the modules to [ ] stainless steel plates. The applicant was requested to explain why these materials were revised, how the new material properties compared to those of previous materials, and demonstrate that the new material properties are equivalent to, or better than, the properties used in the original analysis and design of the AP1000 CIS structures.

In letters, dated February 27, 2009 and July 2, 2009, the applicant identified the use of [ ] as acceptable carbon steel materials for use in the structural modules because these two materials are considered to have equivalent specifications commonly used for rolled shapes. The applicant also explained that the reason for replacing [ ], [ ] stainless steel plates [ ], [ ], for the modules is that [ ] material is not available in the required plate sizes. The staff found that [ ] have substantially different yield strengths, and that the two stainless materials also have different yield strengths. In addition, it is not clear which material was used in the various designs for qualifying the modules. Therefore, in a follow-up RAI, the applicant was requested to demonstrate that the alternative materials are equivalent to, or better than, those used in the original analysis and design of the modules.

In a letter dated August 31, 2009, the applicant provided information that demonstrated that the alternative materials for the structural modules are equivalent to, or better than, those used in the analysis and design. This was demonstrated for both the carbon steel and stainless steel materials, and, therefore, the staff concluded that the proposed use of these new materials is acceptable. The RAI response also provided some markups to reflect this change in the AP1000 DCD. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and table, which resolve this issue.

3.8.3.5 Design Summary Report

In the AP1000 DCD, Revision 16, the applicant revised Section 3.8.3.5.7, "Design Summary Report." The AP1000 DCD, Revision 15 indicated that the results of the evaluation would be documented in an as-built report by the COL applicant. In the AP1000 DCD, Revision 16, this was revised to state that "The results of the evaluation will be documented in an as-built summary report." Thus, the phrase "by the Combined License applicant" was removed. The need to prepare the as-built summary report is being addressed by the applicant as an ITAAC. The staff's evaluation of the need to prepare the as-built report under an ITAAC is discussed in Section 3.8.6, "Combined License Information," in this report.

3.8.3.6 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the applicant's application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 CISs as documented in the AP1000 DCD Revision 19, against the relevant acceptance criteria as listed above and in NUREG-0800 Section 3.8.3.

Based on confirmatory review of the subsequent revision to the AP1000 DCD, the staff finds that the design of the CISs continues to meet all applicable acceptance criteria. In summary, based on the above discussions, the staff finds that the design of the AP1000 CIS is acceptable.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each COL applicant would have to address these issues individually.

3.8.4 Other Seismic Category I Structures

Using the regulatory guidance in NUREG-0800 Section 3.8.4, "Other Seismic Category I Structures," the staff reviewed areas related to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its internal structures. The specific areas of review provided in NUREG-0800 Section 3.8.4 are as follows: (1) description of the structures; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, special construction techniques, and QA; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 Section 3.8.4 and the following SER sections provide the staff's evaluation for the relevant areas. The AP1000 DCD amendment incorporates substantial changes to the shield building design, as well as additional analyses to confirm the adequacy of the design. As a result, this evaluation of the shield building replaces the evaluation in Section 3.8.4.1.1 of NUREG-1793 in its entirety, as well as changes to other portions of Section 3.8.4 relevant to the shield building.

In its previous evaluations of AP1000 DCD Section 3.8.4, the staff identified acceptance criteria based on the design meeting the relevant requirements in 10 CFR 50.55a; 10 CFR Part 50, Appendix A, GDC 1; GDC 2; and GDC 4. The staff found that the design of the AP1000 other seismic Category I structures was in compliance with these requirements, as referenced in NUREG-0800 Section 3.8.4 and determined that the design of the AP1000's other seismic

Category I structures, as documented in the AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

In the AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.4 of the certified design:

1. As a result of the extension of the AP1000 design from just hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the NI structures were performed. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the re-analyses included the additional use of response spectrum and time history methods of analysis. Appendix G of DCD Revision 17 indicates that the response spectrum analysis was used for the 3D refined finite element model of the NI and for the analysis of the PCS valve room and miscellaneous - steel frame structures, flexible walls, and floors. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

2. In DCD Revisions 16 and 17, the applicant revised the design and analysis procedures under Section 3.8.4.4.1 - Seismic Category I Structures. In particular, this section was revised significantly to reflect the change in the design of the shield building.

3. In DCD Revision 16, the applicant revised Section 3.8.4.5.3 - Design Summary Report. DCD Revision 15 indicated that the results of the evaluation will be documented in an as-built summary report by the Combined License applicant. In DCD Revision 16, this was revised to state that “The results of the evaluation will be documented in an as-built summary report.” Thus the phrase “by the Combined License applicant” was removed.

4. In DCD Revision 16 and 17, the applicant revised Section 3.8.4.6.1.1 - Concrete, regarding the concrete material. For the shield building structure, the compressive strength of concrete was increased from 4,000 to 6,000 psi.

3.8.4.1 Description of Other Seismic Category I Structures

During the review of the AP1000 DCD Tier 2, Revision 16, the staff identified that several revisions were made to AP1000 DCD Section 3.8.4.4.1 and Appendix 3H, some of which are Tier 2* information. In RAI-SRP3.8.4-SEB1-03, the staff requested that the applicant explain why these revisions have been made, demonstrate the design adequacy of these changes, and justify the removal of design information from the AP1000 DCD.

In a letter dated May 4, 2009, the applicant provided explanations of why changes were made in AP1000 DCD Section 3.8.4.4.1 and Appendix 3H. The applicant indicated that these are due to design changes to address the, “enhanced shield building design” features and these changes were already communicated to the NRC in APP-GW-GLR-045, Revision 1, which was later revised again to Revision 2. In a letter dated March 5, 2010, the applicant provided mark-ups to Appendix 3H of the AP1000 DCD, which restore some of the design information that was previously removed. The staff found that the restored information was not complete regarding identification of the required reinforcement for concrete sections, reduction in the number of critical sections evaluated, why certain loads do not appear in the load combinations, and apparent inconsistency in the allowable stress values. Therefore, in a follow-up RAI, the applicant was requested to address these items. In addition, there were a number of issues still outstanding with the changes related to the enhanced shield building design and the removal of Tier 2* information.

In response to the above requests, the applicant's letters dated July 26, 2010, and August 30, 2010, provided proposed mark-ups to AP1000 DCD, Appendix 3H, which: (1) add to the corresponding tables the required reinforcement for concrete sections and an appropriate number of critical sections evaluated; (2) present a revised table that incorporates the design changes related to the enhanced shield building design; and (3) propose two new criteria, the same as presented in the evaluation for the response to RAI-SRP3.8.3-SEB1-05 in this SER, for identifying when changes in the values presented in the critical section Tier 2* tables must be reported to the NRC. In addition, the responses also explained that certain loads in some load combinations were excluded because the loads were not applicable to that load combination or that load combination did not govern the design. The differences in some of the tabulated allowable stress values are due to differences in the stress limit coefficients for tension and compression. The staff's review of the responses concluded that they are acceptable, in part, because: (1) corrections were made to include the required reinforcement for concrete sections and an adequate number of critical sections were evaluated; (2) the critical section table was updated to reflect the design changes related to the enhanced shield building design; and (3) explanations were provided to justify why certain loads do not need to be considered.

Tier 2* information is part of the safety analysis report that cannot be changed by a license holder without prior approval. However, the criteria for identifying when changes in values presented in the critical section Tier 2* tables do not have to be reported to the NRC are not in compliance with the regulatory requirements of 10 CFR 52, Appendix D, Section VIII.6.a, because: (1) any changes made to the Tier 2* italicized or bracketed and asterisked text require prior NRC approval; and (2) a generic criterion whereby changes in the loads or member forces that result in an increase in the required reinforcement (or plate thickness for modules) greater than 10 percent also need to be reported. The key is that the required reinforcement or plate thickness cannot change because if the Tier 2* information changes then criterion number (1) applies and it must receive prior approval from the NRC. It should be noted that the proposed criteria for Tier 2* also apply to AP1000 DCD Section 3.8.5.4.4, Table 3.8.5-3, and AP1000 DCD Appendix 3H, for which the applicant also plans to use the new criteria. The staff met with the applicant on October 14, 2010, to provide this feedback. As a result, the applicant stated it was withdrawing TR-57 by letter dated October 21, 2010, and revising the DCD to include revised Tier 2* information in Revision 4 to the response to RAI-SRP3.8.4-SEB1-03, dated October 21, 2010. In this response, the applicant included new criteria on Tier 2* items in Subsection 3H.1 to be consistent with American Society for Testing and Materials (ASTM)-6, "Standard Specification for General Requirements for Rolled Structural Steel Bars, Plates, Shapes, and Steel Piling," and ASTM-A480, "Standard Specification for General Requirements for Flat-rolled Stainless and Heat-Resisting Steel Plate, Sheet, and Strip." In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and tables, which resolve this issue.

<u>Nuclear Island Evaluation of Critical Sections Under Technical Report TR-57 and Report APP-1200-S3R-003</u>

The applicant submitted versions of TR-57 on November 21, 2007, and July 1, 2008, to summarize the structural design and analysis of structures identified as "Critical Sections" in the CIS, auxiliary building, and enhanced shield building. The design of the critical sections for the

CIS is summarized in AP1000 DCD Section 3.8.3. The design of the critical sections for the auxiliary and shield building is described in AP1000 DCD Appendix 3H, Section 3H.5. Two of the critical sections identified in Section 3H.5 are not included in Revision 0 of TR-57. According to TR-57, Revision 0, the information on the evaluation of these two sections will be provided in an update to TR-57 when the security-related assessment is more complete. Further, the information in TR-57, Revision 0, represents the results of detailed calculations and analyses. According to the TR-57, Revision 0, the results will change slightly during the design finalization due to anticipated spectra changes resulting from resolution of the high frequency issues and plant security issues. TR-57, Revision 0, also states that small changes in modeling and updates to software may also have a minor effect on the results.

Subsequently, the applicant made further revisions to the shield building design and analyses, and submitted Revision 1 to the report. This report was later revised and completed in Revision 2, transmitted to the NRC in a letter dated July 1, 2008. TR-57, Revision 2, provides the design of five critical sections for the CIS and 12 critical sections for the auxiliary building. A brief description of the design of two critical sections associated with the enhanced shield building design is also presented. For comparison, the AP1000 DCD, Revision 17, as well as the certified design in the AP1000 DCD, Revision 15, also identifies the same critical sections for the CIS and auxiliary buildings.

In addition to TR-57, the applicant also submitted for the staff's review APP-1200-S3R-003, Revision 0, "Design for the AP1000 Enhanced Shield Building," dated August 31, 2009. The purpose of this document was to provide a separate report, which specifically describes the enhanced shield building design methodology, testing, constructability, and inspection. The enhanced shield building report includes the design of three regions/locations: shield building cylinder; shield building roof, exterior wall of the PCS water storage tank; and shield building roof, tension ring, and air inlets.

The NRC sent a letter, dated October 15, 2009, to the applicant on the results of its review of the applicant's August 31, 2009 design methodology report for the AP1000 shield building. The letter stated:

> By letter dated August 31, 2009, the applicant submitted its design methodology report for the AP1000 shield building. The U.S NRC has completed its review of that report. Based on that report and the body of technical information reviewed to date, the NRC has determined that the proposed design of the shield building will require modifications in some specific areas to ensure its ability to perform its safety function under design basis loading conditions and to support a finding that it will meet applicable regulations (i.e., 10 CFR 50.55a and 10 CFR Part 50, Appendix A (GDC 1 and 2)).
>
> Specifically, the design of the steel and concrete composite structural module (SC module) must demonstrate the ability to function as a unit during design basis events; the design of the connection of the SC module to the reinforced concrete wall sections of the shield building must demonstrate the ability to function during design basis events; the design of the shield building tension ring girder, which anchors the shield building roof to the wall, must be supported by either a confirmation test or a validated (or benchmarked) analysis method.

During the review of the August 31, 2009 report, the staff identified a potential error in the applicant's computer code, which had been used to proportion the cross-sectional strength of members involving concrete materials (basemat, CIS, auxiliary building, and the shield building). The staff informed the applicant about this concern and the staff's evaluation of the resolution for this issue is described in Section 3.8.5 of this report, regarding the basemat, where this item is identified in RAI-TR85-SEB1-29.

In a meeting held on November 18, 2009, with the applicant to discuss its new proposal on the design of its shield building, the staff indicated that the applicant did not appear to have implemented the 100-40-40 method for combination of the three direction seismic loading in accordance with RG 1.92, Revision 2, or the ASCE 4-98 method. The implementation of the 100-40-40 combination method is also discussed in Section 3.8.5 of this report, regarding the basemat, where this item is identified in RAI-TR85-SEB1-27.

To address the various issues related to the use of the SC module in the shield building and the design of the connection of the SC module to the RC sections, the applicant performed additional analyses and testing and submitted a revised shield building report to the staff for review. Revision 3 to the shield building report was submitted by letter dated September 20, 2010.

The staff's evaluation and acceptance of the design of the critical sections in TR-57, as provided under the AP1000 DCD, Revision 15, was presented in NUREG-1793. However, because of changes in the design of the shield building, the number of critical sections has increased. The staff's review of the additional critical sections associated with the shield building is provided in Section 3.8.4.1.1 of this report. In a letter dated October 21, 2010, the applicant clarified the design basis for the proposed facility by deleting TR-57 and removing references to TR-57 from the DCD.

New Fuel Racks and Spent Fuel Racks - Technical Reports: TR-44 and TR-54

The applicant submitted TR-44, Revision 0, to summarize the structural/seismic analysis of the AP1000 new fuel storage racks. In addition, the applicant submitted TR-54, Revision 0, to summarize the structural/seismic analysis of the AP1000 spent fuel storage racks. Subsequently, additional revisions were made to these TRs to incorporate changes made in response to RAIs regarding the structural analysis and design of the new and spent fuel racks for various loads and in response to related discussions held during several past design audits.

Section 3.8.4 of AP1000 DCD, Revisions 16 and 17 indicates that the new fuel and spent fuel storage racks are described in Section 9.1 of the AP1000 DCD. Therefore, a description of the technical information presented in the TRs and the staff's evaluation of the information in these reports are presented in Section 9.1 of this report. The description; applicable codes, standards, and specifications; loads and load combinations; analysis and design approach; acceptance criteria; and construction of the fuel racks are evaluated in Section 9.1 of this report, in accordance with the requirements of NUREG-0800 Section 3.8.4, Revision 2, Appendix D. Some of the key outstanding issues that were identified by the staff and evaluated in Section 9.1 of this SER include acceptable methods for evaluation of the horizontal impact forces at the top of the racks and evaluation of buckling at the bottom of the racks during liftoff caused by the seismic loading. In addition, reconciliation of the new seismic loads from the applicant's SSI reanalysis was needed.

Another issue is the evaluation of the spent fuel rack impact forces on the spent fuel pool walls. The concern is that with the reanalysis of the spent fuel racks to incorporate the updated seismic loading and revisions in the design of the racks the maximum impact force from a spent fuel rack onto the pool walls increased substantially. This issue is captured under RAI-SRP9.1.2-SEB1-06. In response to this RAI, the applicant's letter dated August 25, 2010, addressed the remaining questions regarding this issue. This response is also evaluated under Section 9.1.2 of this report.

Design Summary Report

In the AP1000 DCD, Revision 16, the applicant revised Section 3.8.4.5.3, "Design Summary Report." The AP1000 DCD, Revision 15 indicated that the results of the evaluation would be documented in an as-built report by the COL applicant. In the AP1000 DCD, Revision 16, this was revised to state, "The results of the evaluation will be documented in an as-built summary report." Thus, the phrase, "by the Combined License applicant," was removed. Preparation of the as-built summary report is being addressed by the applicant as an ITAAC. The staff's evaluation of the need to prepare the as-built report under an ITAAC is discussed in Section 3.8.6, "Combined License Information," in this report.

3.8.4.1.1 Shield Building

The applicant applied for an amendment to the certified design of the AP1000, an advanced, passive, pressurized-water reactor (PWR) design. The staff has reviewed the revised design of AP1000 seismic Category I structures, including the shield building, as described in Revision 17 of the DCD. The staff applied the guidance provided in Section 3.8.4, "Other Seismic Category I Structures," Revision 3, issued May 2010, of NUREG-0800.

This evaluation of the shield building is based on key design-specific issues. These issues are outlined in NUREG-0800: (1) description of the structures; (2) applicable codes, standards, and specifications; (3) loads and loading combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, special construction techniques, and QA; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions.

The staff issued NUREG-1793 in September 2004 and Supplement 1 in September 2005. Revision 15 of the AP1000 DCD was incorporated into Appendix D to 10 CFR Part 52. Subsequently, the applicant submitted Revisions 16 and 17 to the AP1000 DCD with additional modifications to the TRs that relate to the shield building:

- APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," dated May 7, 2010 (Shield Building Report)

- TR-85

- TR-03

With these revisions, the applicant is seeking to make the changes discussed below specific to the design of the shield building.

3.6.1.1.1.1 Safety Function and Description of the Shield Building

The shield building is a safety-related seismic Category I structure that provides structural and radiological shielding for the CV and radioactive systems located in the containment building; protects the containment from external events, including missiles, tornadoes, and seismic events; provides radiation shielding from nuclear materials in containment; supports the PCCWST; and provides for natural air circulation cooling for the CV.

The staff notes that the design of the shield building in the AP1000 is unique in that it is the first shield building design to include the support of the PCCWST at the apex of the building structure. The PCCWST holds 3.039×10^6 kg (6.7 million pounds) of emergency cooling water. This water load accounts for a considerable portion of the load on the roof of the shield building.

The shield building consists of cylindrical walls surrounding, and set at a distance from, the steel containment and a conical roof that supports the PCCWST over the containment. The cylindrical wall of the shield building supports both the roof and the PCCWST. The shield building wall is constructed with both conventional RC and new, first-of-a-kind SC wall modules, which make up about 75 percent of the structure. The SC modules consist of two steel faceplates and have concrete filled in between the faceplates. Shear studs anchor the concrete to the steel faceplates, and tie-bars connect the two outer faceplates together. The shield building roof, an RC structure, is connected to the cylindrical wall by the ring girder/tension ring. The auxiliary building roof and the external walls are connected to the SC cylindrical portion of the shield building. The floor slabs and interior structural walls of the auxiliary building are also structurally connected to the RC cylindrical portion of the shield building. The SC wall is attached to the top and sides of the RC wall with stepped and asymmetrical boundary conditions both in the vertical (meridional) and horizontal (hoop) directions (Shield Building Report, Figure 3.2-2). The SC module steel faceplates are not directly anchored to the RC walls. The SC wall and the RC wall are connected through mechanical connectors (Shield Building Report, Figures 4.1-2, 4.1-3, 4.1-4, and 4.1-5), and the SC wall is also connected to the basemat reinforcement through mechanical splices.

The shield building structure has the following main features:

- a cylindrically shaped wall constructed of SC modules that are stacked vertically, welded together to form a cylinder, and filled with concrete
- an air-inlet region located above the cylindrical wall, designed to allow air flow for containment cooling during certain design basis accidents
- a conical RC roof structure with an integral RC water tank, called the PCCWST. The PCCWST contains approximately 6.7 million pounds of water.
- a ring girder tension ring consisting of a steel box girder filled with concrete, located at the intersection of the conical roof and the air-inlet region
- mechanical connections where the SC wall joins the RC wall

<u>Cylindrical Wall</u>. The executive summary of the Shield Building Report describes the cylindrical SC wall. Figure ES-3 shows the SC wall panel layout, [ ]. The thickness of the SC wall for the air-inlet region varies from [ ].

The free-standing vertical span of the west wall, the height from the top of the basemat to the bottom of the tension ring, is 50.6 m (166 ft, 3 in). The east part of the SC wall connects to the RC wall of the shield building (the part of the 0.9 m (3 ft) thick wall protected by the auxiliary building structure) below the roof of the auxiliary building at El. 44.8 m (146 ft 10 in). The RC floors and walls of the auxiliary building are connected to the RC wall of the shield building and constrain lateral displacement of this wall. The height of the east wall above its SC/RC connection located below the roof of the auxiliary building is 36.4 m (119 ft, 5 in).

Air-Inlet Region. The air-inlet region at the top of the cylindrical wall of the AP1000 shield building has through-wall openings for air flow. These air-inlet openings consist of [ ] steel pipes at a downward inclination [ ] from the vertical. Center-to-center horizontal spacing of these tubes is [ ]. The air-inlet pipes are welded to the steel faceplates. Welded steel studs connect the steel pipes to the concrete.

Roof and PCCWST. The AP1000 shield building roof is a conical RC structure supported by a steel frame consisting of radial steel beams (main roof beams). Metal studs connect a steel plate to the bottom face of the conical RC roof slab. Two vertical, concentric RC walls on the roof, integral with the roof structure, define the boundaries of the PCCWST. At the center of the PCCWST on the roof is an air diffuser, or chimney, that is defined by the inner PCCWST wall.

Tension Ring. The main component of the tension ring is a rectangular, concrete-filled, closed section built of [ ] thick welded steel plates. At the top of the tension ring is a concrete-filled, triangular, closed section of steel plates. The bottom plate of this triangular section is the top plate of the tension ring. The exterior top plate of the triangular section is parallel to the roof slope, while the other top plate is perpendicular to the roof slope to support the roof slab and to anchor some of the roof's reinforcing bars. Attached to the tension ring are interior beam seats that support the radial roof framing girders. Steel plates stiffen the tension ring where these beams are seated.

SC/RC Connections. The SC wall of the shield building connects to the top of the RC basemat (El. 30.5 m (100 ft)) at the bottom of the west wall (for a span of 152.97 degrees). A short portion of the horizontal west wall connection, between azimuths 175.63 degrees and 190.00 degrees, is at El. 33.2 m (109 ft) with a vertical connection at azimuth 190.00 degrees at the transition between El. 30.5 m (100 ft) and El. 33.2 m (109 ft). The east part of the SC wall has a horizontal connection to the RC wall of the shield building below the roof of the auxiliary building at El. 44.8 m (146 ft 10 in), and vertical connections to the sides of this RC wall at azimuth 341.94 degrees, near Wall Q, from El. 30.5 m (100 ft) to El. 44.8 m (146 ft 10 in), and at azimuth 174.60 degrees, near wall N, from El. 33.2 m (109 ft) to El. 44.8 m (146 ft 10 in).

The staff finds that the description of the shield building structure, as provided in the Shield Building Report and as supplemented with design information in the responses to staff questions at the meeting on June 9-11, 2010, provides sufficient information to define the primary structural aspects and elements used by the applicant to design the structure to withstand the design-basis loads.

Using the guidance described in NUREG-0800 Section 3.8.4 and related RGs, the staff reviewed areas related to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its interior structures.

In its previous evaluation of Section 3.8.4 of the AP1000 DCD in NUREG-1793, the staff accepted the design of these structures because it met the following applicable requirements of 10 CFR Part 50:

- 10 CFR 50.55a
- Appendix A
  - GDC 1
  - GDC 2
  - GDC 4

In Revisions 16 and 17 of the AP1000 DCD, the applicant proposed the following changes to Section 3.8.4 of the certified design:

- As a result of the extension of the AP1000 HR design to a design that includes a broader range of soil profiles, the applicant performed various seismic reanalyses of the NI structures. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the reanalyses included the additional use of response spectrum and time history methods of analysis. Appendix 3G to Chapter 3 of the AP1000 DCD, Revision 17, indicates that the RSA was used for the three-dimensional refined finite element model of the NI and for the analysis of the passive containment cooling water system valve room and miscellaneous steel frame structures, flexible walls, and floors. Time history analyses were used to determine maximum soil bearing pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

- In DCD Revisions 16 and 17, the applicant revised the design and analysis procedures in Section 3.8.4.4.1 regarding seismic Category I structures. In particular, the applicant revised this section significantly to reflect the change in the design of the shield building. The shield building design has evolved as described primarily in the Shield Building Report.

- In DCD Revisions 16 and 17, the applicant revised Section 3.8.4.6.1.1, "Concrete." For the shield building structure, the compressive strength of concrete was increased from 27.58 MPa (4,000 psi) design strength in the RC areas to 41.37 MPa (6,000 psi) design strength in the SC structural modules. The applicant revised the test age of concrete from 28 days to 56 days and changed some details about the chemical composition in the Portland cement and the proportioning of the concrete mix.

- In TR-03, the applicant compared the corresponding acceleration profiles obtained from the SSI analyses for the various soil sites to the original HR acceleration profile used in the design of the AP1000. On the basis of this comparison, the applicant concluded that the AP1000 design is adequate for the range of soil sites considered.

- In response to questions from the staff relating to the above issues (discussed below), the applicant redesigned the shield building based on feedback from the staff transmitted in an NRC letter dated October 15, 2009. The Shield Building Report describes these design changes.

Based on its evaluation of the proposed shield building design provided in Revisions 16 and 17 to the AP1000 DCD, the staff issued RAI-SRP3.8.3-SEB1-01 asking the applicant to provide information about the design methodology and to specify which aspects of the shield building design are in accordance with ACI 349, as modified by the additional criteria in RG 1.142, Revision 2, and ANSI/AISC N690. In a letter dated August 31, 2009, the applicant submitted its design methodology report, APP-1200-S3R-003, Revision 0. In a letter dated October 15, 2009, the staff identified modifications that would be required to ensure that the shield building could perform its safety function under design-basis loading conditions and to support a finding that it meets the applicable regulations in 10 CFR 50.55a and GDC 1 and 2 in Appendix A to 10 CFR Part 50.

The letter identified the following key issues:

Detailing, Design, and Analysis

1. The applicant needs to demonstrate the adequacy of the design and detailing of the SC module to function as a fully composite unit as assumed in the applicant's design/analysis. In addition, the applicant needs to demonstrate that the SC module has sufficient ductility to survive severe earthquakes or tornado winds.

2. The SC module wall to RC wall connection is to be designed and detailed for both the RC and SC portion of the connection and supported by a basis for why the connections will carry the shield building design loads.

3. The design and analysis of the shield building tension ring (i.e., ring girder) and the air-inlet region should be supported by a validated design/analysis method (i.e., benchmarked to experimental data), or by confirmatory model tests.

Based on subsequent interactions, including meetings in December 2009 and January and February 2010, as well as telephone conferences between the NRC and the applicant, the applicant submitted APP-1200-S3R-003, Revision 1. Following the March submittal and after several telephone conferences between the NRC and the applicant, the applicant submitted APP-1200-S3R-003, Revision 2 (the Shield Building Report). The staff reviewed the Shield Building Report and held a public meeting with the applicant on June 9-11, 2010. The meeting resulted in 21 items for applicant action, as summarized in an NRC memorandum dated July 19, 2010. The action items required the applicant to address design methods, analyses, and testing issues to help demonstrate the adequacy of the shield building design.

The applicant responded to 18 action items in its June 30, 2010, submittal and responded to the remaining Action Items 4 and 12 on July 23, 2010, and July 31, 2010. The applicant responded to Action Item 21 on September 3, 2010.

The applicant provided the following information in response to the action items:

- analysis methods, results, and justification for the structural demand and capacity of the shield building

- analysis and results, including stress/strain test data, and analysis of test specimens using material models in [   ]

- justification to support global stability in the design of the structure

- design approach and load path for the SC/RC connection, including justification for the shear friction capacity of the connection and any resulting design changes that were made based on the respective evaluations

- justification and qualification and production criteria for the use of mechanical splices in the design of the SC/RC connection

- analysis to support the design of the ring girder and the connection between the ring girder and air-inlet region of the SC wall, including a comparison of the cross-sectional forces between [ ] and [ ] codes to verify shear friction loads

- analysis to support the adequacy of the [ ] used at the transition of the SC wall at the air inlets from 91.4 cm to 137.2 cm (36 in to 54 in) thickness

- evaluation of the effect of concrete cracking on the structural design

The applicant also submitted a supplemental report, “Final Shield Building In-Plane Shear Test Results,” dated June 24, 2010, on the testing of the SC module under cyclic in-plane shear. Section 3.8.4.1.1.3.5 of this report describes the staff’s evaluation of this test.

3.8.4.1.1.2 Regulatory Basis

The AP1000 shield building protects the reactor and containment from exterior missiles generated by tornadoes and, thus, it is subject to impact loads. The AP1000 shield building is classified as a seismic Category I structure because it should remain functional during severe earthquakes. Therefore, the shield building is subject to both seismic and impact loads and is designed and evaluated in accordance with the regulations and guidance as follows:

- 10 CFR Part 50.55a(a)(1) requires, “safety-related structures, systems, and components be designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with the importance of the safety functions to be performed.”

- GDC 1 states, “Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.”

- GDC 2 states, “Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their functions.”

- NUREG-0800 Section 3.8.4 refers to RG 1.142 and ACI 349.

- RG 1.142 endorses ACI 349 and sections of ACI 318, “Building Code Requirements for Structural Concrete and Commentary,” issued 2008, as applicable codes for all seismic Category I concrete structures, including concrete shield buildings other than containment structures.

3.8.4.1.1.3 Evaluation

This evaluation is limited to the design basis of the shield building and does not address its ability to protect against a malevolent aircraft crash, which is a beyond-design-basis event evaluated under NUREG-0800 Chapter 19, “Severe Accidents.”

3.8.4.1.1.3.1 Design Methodology and Process for Shield Building Design

In response to staff questions regarding the design methodology and the process for the design of the shield building, the applicant summarized its design process in a matrix in Table 1.2-1 and described it in Chapter 2 of the Shield Building Report. According to this description, the concrete design of the following areas of the AP1000 shield building falls directly within the scope of ACI 349:

- shield building roof
- knuckle region of the roof near the PCCWST wall
- compression ring
- PCCWST

The applicant designed these areas in accordance with the provisions in the established design codes by using linear elastic analysis methods. Specifically, the design for the sections in these areas is based on compliance with the ACI 349 Code, as supplemented with guidance in RG 1.142 for concrete structures. The design of the sections in these areas, which uses established design codes and analysis methods listed in Section 3.8.4 of NUREG-0800, satisfies the regulatory basis listed above and is, therefore, acceptable to the staff.

The following other areas of the shield building structure are considered as special features of the design because the applicant used SC modular wall design:

- SC cylindrical wall
- SC/RC connection
- air-inlet region

Codes and standards for the design of SC modular wall and associated structural components do not exist in the United States. Design guidelines for SC modular construction already exist in Japan, namely Japan Electric Association Code, Guideline 4618, “Technical Guidelines for Aseismic Design of Steel Plate Reinforced Concrete Structures—Buildings and Structures,” issued in 2005. However, these guidelines were not specifically developed for external structures with configurations like those of the AP1000 shield building and have not been approved by the NRC. In the Shield Building Report, the applicant designated the areas of the building that use SC modular construction, which include the SC/RC connections, as special structures under ACI 349, Section 1.4.

The applicant applied the provisions of the established ACI 349 Code to the design of these special structures using linear analysis, nonlinear analysis, and testing. Specifically, the applicant applied the provisions in ACI 349 for the design of RC seismic Category I structures to the design of SC wall modules in the AP1000 shield building design. To validate the use of the code, the applicant performed nonlinear analysis and conducted a testing program to verify the behavior and determine the stiffness, strength, and ductility of proposed SC wall modules under monotonic and cyclic loads. In addition, the applicant reviewed international test data on SC wall modules (Appendix A to the Shield Building Report) to confirm the adequacy of the assumptions used by the integrated design process, such as the assumption that the SC wall modules would function as a composite unit under design-basis loads.

The integrated design process for the SC wall module uses standard methods of analysis to calculate stress demands on the shield building that meet the acceptance criteria in NUREG-0800, namely, linear elastic structural analysis. In addition, the design process uses benchmarked nonlinear analysis to confirm that cracking would not cause significant changes in the design demands; that is, changes that would lead to stresses that would invalidate the design obtained with the extension of the established code provisions.

The applicant's integrated design process also makes use of the design process for structural steel components in certain areas of the shield building. Specifically, it uses ANSI/AISC N690 in designing structural steel components of seismic Category I structures. The applicant used ANSI/AISC N690 in designing the following areas of the shield building:

- the steel roof that supports the concrete roof slab
- tension ring
- SC/RC connection

The design process uses provisions from two different design codes: ACI 349 Code for RC components, which uses an ultimate strength design approach and ANSI/AISC N690 Standard for steel and composite components, which uses an allowable stress design approach. The use of two different codes necessitates that the components or parts of components assessed against each code are clearly distinct and that appropriate load combinations are used for each case. The staff's review of the Shield Building Report concludes that these conditions have been met in an acceptable manner.

Based on the discussion above, the staff accepts the applicant's use of the design methodology provided in ANSI/AISC N690 Standard for structural steel components to design the shield building tension ring and the roof supporting steel beams. In addition, the staff accepts the applicant's approach of using ACI 349 as the basis for the design of the other areas, namely the shield building roof, the knuckle region of the roof near the PCCWST wall, the compression ring in the roof, and the PCCWST and walls.

The staff finds that although ACI 349 is not explicitly applicable to the SC modules, the applicant's design method, which is fundamentally based on ACI 349 and supported by confirmatory analysis and testing to confirm the adequacy of the design, is acceptable.

The staff's evaluation of the technical basis, including testing, confirmatory analysis, and design detailing, that supports this integrated design method appears in subsequent sections of this report.

3.8.4.1.1.3.2 Design of the Shield Building

In the Shield Building Report, the applicant made significant design changes from previous versions of the design by replacing lap splices with mechanical splices at the SC-to-RC connection region between faceplates, increasing the thickness of SC module faceplates from [ ], using more ductile steel, and proposing a testing program to include testing for ductility and behavior under cyclic loads. The applicant also replaced the SC tension ring with a steel box girder, redesigned the air-inlet area with fewer through-wall openings, modified the concrete roof design from an SC module design method to an ACI 349 design method, moved SC/RC connections in the east side of the wall downward and away from the original area where the auxiliary building roof connected to the wall in order to avoid congestion and stress concentrations in the area, reduced the use of self-consolidating concrete, and redesigned the SC/RC connection to provide a direct load path. The applicant also replaced the original high-strength smooth anchor rods between the SC-to-RC basemat with #14 mild steel deformed reinforcing bars, as discussed during the meeting of June 9-11, 2010. The staff considers these changes to be significant improvements in the design of the structure to enable it to function as a unit under design-basis loads.

The staff evaluation of the applicant's analysis for the changes is provided below.

Levels of Analysis

The applicant's approach to developing the design basis involves three levels of analysis as described in the Shield Building Report, Section 2.6, Table 2.6-1. The three levels of analysis, with increasing levels of model refinement, are as follows:

Level 1 is used for determining the load magnitudes (seismic demands) imposed on the structure. Level 2 is used for determining the member forces and deformation demands. Level 3 is used to assess the region with high stresses, strains, and displacements in the shield building, such as the connection regions. Linear elastic models are used at Levels 1 and 2. At Level 3, nonlinear analysis is used to confirm the results at the various levels of analysis.

The applicant used the Level 1 analysis to generate the design-basis ISRS and load magnitudes on the AP1000 NI. The applicant used the [ ] NI20 and [ ] NI10 models to develop ISRS and to design and analyze seismic Category I SSCs. In these analyses, the concrete material modulus of elasticity was reduced to 80 percent of its nominal value to account for minor concrete cracking. The applicant performed confirmatory analysis of the Level 1 analysis using the [ ] finite element analysis code. To accomplish this, the [ ] NI20 model was converted to an [ ] model with the capability to account for concrete cracking. The nonlinear concrete material parameters were benchmarked to SC element tests performed at Purdue University. Chapter 8 of the Shield Building Report describes the results of this confirmatory analysis.

The applicant used the Level 2 analysis to calculate structural design demands for the AP1000 NI. These analyses used the [ ] NI05 building model, which has a characteristic element size of 1.5 m (5 ft). In Section 2.6 of the Shield Building Report, the applicant stated that the accuracy of the NI05 model was validated by comparing the dynamic response to the [ ] NI10 model, which has a characteristic element size of 3 m (10 ft). The applicant performed confirmatory analysis of the Level 2 analysis using the [ ] finite element analysis code. The [ ] model is a highly refined model that explicitly accounts for the steel and concrete materials with separate shell and solid elements. In addition, nonlinear properties are used to characterize the concrete and steel materials. In Section 2.6 of the Shield Building Report, the applicant stated that the [ ] code was benchmarked to the Purdue University testing, as described in Chapter 7 of the Shield Building Report.

The applicant performed the Level 3 analysis to determine stresses, strains, and displacements of the critical high-stress regions in the shield building design using the [ ] finite element code and nonlinear inelastic material modeling. The concrete material parameters were benchmarked against Purdue University test results. The detailed submodels used included elements such as concrete, steel plate, studs, and [ ]. A strain-based failure criterion was selected to ensure acceptable limits under design-basis loads. Results from the Level 2 [ ] analyses are "handed-off" to the Level 3 [ ] analyses by imposing displacements at the boundary of the Level 3 analysis. The applicant described this handoff procedure in Appendix C.3 of the Shield Building Report.

The staff finds the design approach involving the three levels of analysis to determine the load magnitudes (seismic demands), the member forces, and deformation demands and including confirmatory analysis, provides a logical, reasonable, and adequate technical approach to developing the shield building design and, therefore, is acceptable.

The staff accepts the various levels of analysis involving the use of increasingly refined models to better determine element behavior under the design-basis seismic loads (SSE). The models reasonably account for material properties, and the resulting strain and stress data are confirmed under the Level 3 analysis, whereby the results from the standard linear elastic analysis models compare reasonably well with the results from the nonlinear models.

The staff finds that the approach is reasonable in that it enables the applicant to gain a better understanding of the behaviors of the structural elements of the design, particularly in the critical high-stressed regions of the structure such as the SC/RC connection. This SER provides the staff's evaluations of the results of this approach under the subsequent sections.

3.8.4.1.1.3.3 Confirmatory Analysis

In Chapter 8 of the Shield Building Report, the applicant described the approach for its benchmarking analysis methods. It should be noted that the applicant's analysis methods were not benchmarked by updating or "tuning" modeling assumptions to match any particular test. Rather, the applicant provided a confirmatory analysis, whereby it used [ ] and [ ] models to predict the behavior of various elements of the SC module and compared those results to those established using the ACI 349 design methods and SC module tests. The staff reviewed the confirmatory analysis used by the applicant to validate the predicted behavior under design-basis loads, as discussed below.

As previously stated, the applicant's design process for the shield building used standard methods of analysis that meet NUREG-0800 acceptance criteria, namely, linear elastic structural analysis, to calculate stress demands on the building. In addition, the design process uses confirmatory nonlinear analysis to confirm that concrete cracking and steel stresses would not cause significant changes in the design demands.

The applicant also described the approach for its confirmatory analysis methods in the September 3, 2010 supplement to the Shield Building Report. The applicant stated that the goal of the confirmatory process was to develop three-dimensional finite element models for SC structures that can be used to further evaluate the behavior and design of the AP1000 shield building. The applicant used the commercial finite element analysis codes [ ] and [ ] to perform the confirmatory analysis. The critical shield building areas (Section 10.2.2 of the Shield Building Report) designed using ACI 349 were modeled using a detailed Level 3 [ ] analysis for confirmatory purposes. These areas include Wall Q (Section C.6), west wall (Section 10.3 and Section C.5), air inlets (Section C.4), and Wall 5. Section 10.3 of the Shield Building Report summarizes the Level 3 analysis results for these four critical areas. Below is a summary of the applicant's confirmatory analysis methods, including development of the [ ] model, verification of the model predictions with experiments, and performance of the pushout and anchorage tests, followed by the staff's evaluation.

[ ] Model Development

The applicant used the commercial finite element analysis code [ ] to perform confirmatory calculations. Detailed [ ] models of several SC test specimens were developed and included important features of these modules, such as shear studs, [ ], steel plate, and concrete infill.

The steel elements were modeled in [ ] with a reduced integration solid element (C3D8R). The use of this solid element results in faster analysis running times. The nonlinear steel material properties were modeled using a multiaxial plasticity theory with von Mises yield surface, associated flow rule, and isotropic hardening. Table 8.2-1 of the Shield Building Report provides nominal and material parameters for the steel elements for use in the Level 3 analyses. The applicant used measured material properties for the test specimens, described in Chapter 8 of the Shield Building Report.

The applicant modeled the concrete infill using C3D8R elements and a concrete damage plasticity model. This model has isotropic damage rules and can be used for modeling concrete behavior under uniaxial (compression, tension, and shear), cyclic, and multiaxial loading conditions. This model uses a compression yield surface with non-associated flow in compression. In tension, the model uses damaged elasticity concepts to model smeared cracking. The postcracking behavior depends on the tension stiffening modeling used for the concrete. The applicant analyzed three tension-stiffening models: a stress-displacement model (Figure 8.6-3) and two stress-strain models (Figure 8.6-4). As a result of the confirmatory analysis, the applicant selected the stress-strain model in Figure 8.6-4 with the lowest concrete tensile strength for the Level 3 analyses.

The applicant modeled the steel [ ] elements as fully embedded into the concrete infill and verified the approach using pushout tests. Section 8.9 of the Shield Building Report describes the results of these tests. The applicant also conducted finite element mesh sensitivity studies to confirm the adequacy of element size.

In the applicant's supplement to the Shield Building Report dated August 24, 2010, the applicant stated that a limitation of the confirmatory approach is that fracture of steel SC components (e.g., plates, studs, and [ ]) is not explicitly modeled. The applicant chose to establish acceptance criteria (strain limits), based on the guidelines in Nuclear Energy Institute (NEI) 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs" and the applicant's experimental results, for use in analysis as discussed in Section 10.1 of the Shield Building Report. Once the strains in these components exceeded these limits, the analysis results were judged to be no longer valid. In Section 10.1 of the Shield Building Report, the strain limits for steel studs were set at 5 percent while those for reinforcing bars, including [ ] and steel plates, were set at 2 percent. Subsequently, the applicant revised the strain limits on the [ ] to 1.5 percent, as noted in its August 24, 2010 supplement.

Section 10.1 of the Shield Building Report states that the tensile strain limits for the steel faceplates, 2-percent maximum membrane tensile strain, and for the steel reinforcing bars, 2-percent tensile strain, were taken to be half as large as those in NEI 07-13. Tensile strain limits in NEI 07-13 are already set to be conservatively less than the fracture tensile strain limits for steel materials. For the [ ], the final tensile strain limit chosen by the applicant, 1.5-percent strain, is also less than the [ ] strains at maximum tensile stresses shown in response to Action Item 5. The staff has proposed accepting, through DG-1176, “Guidance for the Assessment of Beyond-Design Basis Aircraft Impacts,” issued July 2009, the ductile material strain limits in Table 3-2 of NEI 07-13 for use in aircraft impact analyses. The staff’s review of the applicant’s material strain limits for steel faceplates (2 percent and [ ] (1.5 percent) finds that these limits are more conservative than those in NEI 07-13 (5 percent for SA 516 plate and 5 percent for Grade 60 reinforcing steel). Based on the conservative use of the failure criteria recommended in NEI 07-13, the staff finds the strain limits chosen by the applicant for the steel faceplates and reinforcing bars to be acceptable for use in confirmatory analysis.

For the shear connectors (studs), the applicant set the strain limit at 5 percent for the ASTM A108 Nelson studs. The staff reviewed the Nelson stud material specifications for similar studs and finds that the specifications require a minimum percentage of elongation (5.1 cm (2 in) gage length) of 20 percent for mild steel and concrete anchors. Therefore, the applicant’s use of a strain limit of 5 percent is conservative, based on a comparison to 20-percent elongation over a 5.1 cm (2 in) gauge length. On the basis of conservative use of a failure strain of 5 percent, the staff finds that a strain limit of 5 percent for A108 Nelson studs is acceptable for use in confirmatory analysis.

<u>Verification with Experiments</u>

In its letter dated August 24, 2010, the applicant stated that the modeling approach would be verified by qualitative and quantitative comparisons with experimental observation and results from large-scale tests conducted by the model developers themselves. The applicant compared the predicted shapes, rotations, and cracking pattern with those observed experimentally. The predictions were also evaluated for behavior by comparing the predicted cracking patterns, steel strains, and particularly the mode of failure with those observed experimentally. The applicant also made quantitative evaluations by comparing the predicted load-deformation responses with those measured experimentally.

As an example, the applicant showed the predicted behavior and failure mode for an out-of-plane shear specimen (a/d=3.5)[1] in Figure 2. The applicant stated that the model predicted the location and orientation of concrete cracks, the formation of concrete compressive struts between cracks, and the tensile stresses and yielding of [ ] at the crack locations.

In Figures 3 and 4, the applicant also compared predicted and measured load with midspan displacement response for two out-of-plane shear critical tests (a/d=3.5 and 2.5). The applicant stated that the model predicted the initial and postcracking stiffness with reasonable accuracy and that overall strength and failure were conservatively predicted. The applicant indicated that the models predict tie-bar plastic strains of 1.5 percent, the strain limit for these bars, at a displacement that approximately corresponds to the displacement in the test when the test specimens failed in a brittle manner. Using the above strain limits, the applicant stated that the finite element models were able to predict the behavior of SC modules in the elastic and postcracked regions of response (typically corresponding to load levels up to and beyond the SSE) with reasonable accuracy.

[1] a/d refers to the length of spans to their depth, also referred to as shear span ratio.

In reviewing the applicant's confirmatory analysis, the staff identified several concerns that were discussed at a June 9-11, 2010, meeting and resulted in action items for the applicant related to the analysis benchmarking and methodology:

- In Action Item 12, the staff asked the applicant to provide a typical load case at the SSE level and compare cross-sectional forces for both the standard [ ] Level 1 analysis and for a linear analysis with the [ ] Level 2 model.
- In Action Item 15, the staff asked the applicant to indicate the locations in the calculated load deflection curves where the 2-percent limiting strains (total strains) would occur.
- In Action Item 16, the staff asked the applicant to provide the benchmarking analysis for the [ ] models.
- In Action Item 17, the staff asked the applicant to describe the handoff procedure from the Level 2 model [ ] to the Level 3 model [ ].

The applicant responded to the above action items in its letter dated August 3, 2010. In response to Action Item 12, the applicant compared forces and moments resulting from linear analysis with the [ ] and [ ] models. Both of the models used linear material properties. Table 12-1 of the response compares the forces and moments generated by the two models based on seismic loading at the same location. Based on its review of the results in Table 12-1, the staff finds that the percentage difference in analysis results between [ ] and [ ] is less than 6 percent for axial tension (F_y) and bending moment (M_z). Therefore, based on the applicant's comparison of the results from linear analysis with [ ] and [ ], which indicates a difference of less than 6 percent for the significant cross-sectional forces, the staff finds the applicant's response to Action Item 12 to be acceptable.

In response to Action Item 15, the applicant provided load-deflection plots in Figures 4.1.1-1 and 4.1.1-2 for out-of-plane test specimens with a/d=3.5 and a/d=2.5, respectively. The plots have markings that show the location in the force-displacement curves where plastic strains of 1.5 percent and 2.0 percent occurred in the analysis with the benchmarked models.

In Figure 3-3 of its September 3, 2010, submittal, the applicant compared the maximum out-of-plane shear demand at the design-basis seismic load (SSE level) with test results (a/d=2.5) and analysis prediction. The staff reviewed the force-deflection plots and finds that comparisons of analysis and testing for the out-of-plane specimens (a/d=3.5 and 2.5) agree reasonably well with respect to stiffness for demands up to the SSE level. Based on this observation, the staff finds the applicant's response to Action Item 15 acceptable.

In response to Action Item 16, the applicant provided additional information on the benchmarking of the [ ] model. For in-plane shear on SC modules, the applicant developed a model with the same characteristics as those used in the shield building wall. The inner and outer steel plates were modeled with 0.9 m (3 ft) elements and had a thickness of 1.9 cm (0.75 in). The applicant used the [ ] Winfrith material model and modeled the steel plate with a piecewise linear plasticity model. The model was loaded in pure shear, and the applicant verified the results against scaled Japanese test data (page 111)[2]. The applicant

[2] Westinghouse Electric Company, "Presentation and Actions for WEC Meeting with NRC June 9 through June 11," June 30, 2010. (ADAMS Accession No. ML101940046)

found that the model prediction of the in-plane shear capacity was in good agreement with the expected value.

For out-of-plane shear, the applicant performed additional [ ] confirmatory analysis. The applicant used models that had the same number of elements through the thickness of the wall as that used in the [ ] Level 2 analyses. Results of these comparisons, shown in Table 3.1 of the response to Action Item 16, indicate that the [ ] models are reasonably accurate for SSE load levels as well as for the range of applicability of the [ ] Level 3 models.

For the Level 2 and 3 local models, the applicant provided an example comparison of analysis predictions for the Wall 5 location. The results appear in Figures 4.1.2-27 through 4.1.2-29. The staff's review of these figures finds that the [ ] Level 2 and [ ] Level 3 models compare well for in-plane shear, out-of-plane shear, and axial tension. Based on the applicant's submittal of the [ ] benchmarking analysis, which presented benchmarking results for in-plane, out-of-plane, and Level 2 versus Level 3 models, the staff finds the applicant's response to Action Item 16 acceptable.

In response to Action Item 17, the applicant provided the steps performed to transfer analysis results from the [ ] Level 2 analysis to the [ ] Level 3 analysis, as well as the benchmarking of that procedure. The Level 2 and 3 integrated analysis includes the following steps:

(A) Identify critical regions in the shield building at the RC/SC interface and air-inlet regions.

(B) Generate the Level 2 model of the NI and shield building for the pushover confirmatory analysis, which includes models for the critical regions.

(C) Create Level 3 models for the same regions using the same cut boundary condition as in the Level 2 model.

(D) Perform the Level 2 analysis ([ ]) and extract the displacements at the cut boundaries of the critical regions.

(E) Apply the Level 2 displacements to the corresponding boundaries of the Level 3 models via shell elements that allow the coarse mesh Level 2 displacements to be interpolated and applied to the Level 3 nodes at the cut boundaries.

(F) Analyze the Level 3 models under the applied displacement boundary conditions in step (E).

To verify the adequacy of using displacements at the cut boundaries to transfer results from the Level 2 analysis to the Level 3 analysis, the applicant organized the benchmarking of this transfer method in two parts. The first part of this confirmatory analysis consisted of the following steps:

(A) Generate separate Level 2 models of the critical regions that match those for the Level 2 pushover analysis.

(B) Create Level 3 models for the same regions using the same cut boundary condition as in the Level 2 model.

(C) Apply unit loads at the boundaries of the Level 3 models to determine the stiffness of the Level 3 models for various loadings.

(D) Apply the same unit loads to the corresponding boundaries of the Level 2 models being benchmarked.

With this confirmatory analysis, the applicant assessed the relative stiffness of the Level 2 and Level 3 models. The range over which the response curves under the applied unit loads calculated with both models approximate each other identifies the range over which the two models have similar stiffness and, therefore, the range of acceptability of the handoff procedure. The applicant provided results from the confirmatory analyses in Figures 4.1.3-27 to 4.1.3-29 for Wall 5 and in Figure 4.1.3-31 for the air-inlet region. Based on the results in these figures, the staff finds that the applicant's handoff is acceptable for loads up to the SSE load level.

For the second part of the confirmatory analysis, the applicant developed an example simple shear wall model. The shear wall was loaded with three different loading cases (tension, in-plane shear, and out-of-plane bending) to verify the handoff procedure in different loading scenarios. Comparisons for axial tension (Figure 4.1.3-10), in-plane shear (Figure 4.1.3-12), and out-of-plane bending (Figure 4.1.3-15) show that the model and submodel compare reasonably well. Based on the review of the applicant's description of steps performed to transfer analysis results from the [ ] model and the verification results, the staff finds the applicant's response to Action Item 17 is acceptable.

<u>Pushout Tests</u>

The applicant performed pushout tests to evaluate the interaction between the [ ] that are welded to the steel plates and embedded in concrete infill. In Section 8.9 of the Shield Building Report, the applicant described the approach to conduct the confirmatory analysis for [ ]. All specimens used a [ ] pitch for stud spacing. Specimen 1 used normal concrete with two studs at [ ] spacing on each face with tie-bars in between the studs, while Specimen 2 used normal weight concrete with [ ] at [ ] spacing. Specimen 3 used self-consolidating concrete with [ ] aggregate and [ ]. Figures 8.9-4, 8.9-9, and 8.9-14 compare the analysis results (load displacement) and testing.

In Section 8.9.4 of the Shield Building Report, the applicant described the approach for modeling the [ ], as well as an evaluation of the mesh refinement. The applicant used the embedded method with [ ] concrete and shear connector elements for its simplicity and ability to capture the primary features of the load-slip displacement behavior.

The staff reviewed the applicant's analysis and testing, which provided results for the interactions between the [ ]. The staff reviewed Figure 8.9-4 and finds the applicant's recommended element size of [ ] to be acceptable for confirmatory analysis because the initial stiffness and strength of the shear connectors have a reasonable correlation to the test results.

Anchorage Test

In the Shield Building Report, the applicant performed a confirmatory analysis of an anchorage test. Although the anchorage test design represented an earlier design concept, described in Revision 1 to APP-1200-S3R-003, the applicant felt that the comparison was still useful for confirmatory purposes. The applicant modeled the full-scale test specimen using [ ] and the concrete damage plasticity model. The mesh size for both the [ ] and the concrete elements was 3.8 cm (1.5 in). In Figure 8.10-2 of the Shield Building Report, the applicant provided a comparison of analysis and test results that shows that the Level 3 models predict reasonably well the strains in the steel faceplates and in the dowels for strains up to about 2 percent. Analysis results in Figure 8.10-6 show the location and orientation of concrete cracks and the formation of compressive struts between cracks, which provide a reasonable explanation for the observed behavior under the monotonic load conditions for the test. The staff reviewed the applicant's comparison of test results and analysis predictions and finds that the analysis results agree reasonably for the entire range of response analyzed and for the monotonic load conditions of the test. The staff notes that although the results reflect the early anchorage design, the comparison between the analysis and the test is acceptable for confirming the strains of the faceplates and the dowels. This finding only applies to the benchmarking of the finite element model for monotonic loading. The assessment of anchorage design may be found in Section 3.8.4 of this evaluation.

Confirmatory Analysis Results

Tables 10.3-2 through 10.3-5 of the Shield Building Report provide the results of the confirmatory analysis for critical areas: the air inlets, west wall, Wall Q, and Wall 5. For SSE load levels, the stress levels in the steel plates, [ ] are below the yield level for each component in the west wall, Wall Q, and Wall 5. In the air-inlet region, there is some predicted yielding of studs with a strain of 0.52 percent. However, this strain is less than the assumed failure strain of 5 percent. The staff finds that these results indicate that while there is some degree of concrete cracking predicted by the nonlinear analysis, as expected, the stresses and strains in the shield building critical areas are below yield, with the exception of some local stud yielding in the air-inlet region.

Conclusion on Confirmatory Analysis

In summary, the staff concludes that the applicant has: (1) performed testing to obtain data on the response and behavior for key failure modes of the SC wall modules; (2) developed confirmatory analysis models; (3) shown that the models predict the observed experimental behavior and response with acceptable accuracy up to the design-basis seismic load level (SSE); and (4) used the confirmatory analysis to predict stresses and strains in critical areas of the shield building for the SSE load level. Further, the staff finds that the applicant has adequately addressed the staff's concerns raised in Action Items 12, 15, 16, and 17, as identified in applicant's June 30, 2010, submittal.

Based on the above findings and the applicant's SSE load level predictions of low stress and strain values in the SC steel plates, [ ] the staff finds the applicant's confirmatory analysis approach to be acceptable. Further, the staff finds the applicant's use of the ACI 349 Code for the design of these critical sections to be acceptable.

3.8.4.1.1.3.4 Seismic Demand and Analysis Methods

Chapter 10 of the Shield Building Report describes the applicant's analyses to determine how the seismic demand that is imposed on the AP1000 NI is implemented in the design of the shield building.

The applicant used three-dimensional finite element models generated with the [ ] and [ ] codes to perform the dynamic analyses. These models comprised shell, beam, and solid elements to represent the structural geometry of the NI. For determining the design-basis FRS and demands used for structural design of the shield building, the applicant used the [ ] NI20 model to perform SSI analyses (for soil sites) and the [ ] NI10 model to analyze the HR site condition. Both models idealized the shield building wall structure with a single shell element representing the SC wall module. The staff reviewed this assumption and found it to be unsubstantiated in both TR-03 and in Revision 1 to APP-1200-S3R-003.

The staff was concerned that a single shell element would not be adequate to analyze the complex through-thickness strain gradients expected near structural discontinuities and to account for concrete cracking. Discussed below is the staff's evaluation of the applicant's method of designing the specific components of the tension ring, air-inlet region, W36 beams, conical roof, and PCCWST.

<u>Determination of Responses to Earthquake Loads</u>

For the design of the shield building, the applicant used response spectrum analyses and the [ ] NI05 model to perform seismic analyses. The applicant validated the [ ] NI05 model, which is a refined version of the [ ] NI10 model, against the NI10 model by comparing the mass participation by frequency of the various response modes of the structure. The NI05 model consists of a combination of shell elements, namely [ ] SHELL 45 for most of the SC wall, solid elements, beam elements, and lumped masses to represent the principal components and structures in the NI. The chosen finite elements for the SC modular wall and the overall refinement of the finite element model are adequate for the calculation of design load demands for the shield building wall for a structure with the proportions of the shield building. The input response spectra at the underside of the basemat were determined from the envelope of the response spectra for all soil cases as well as the HR case. The staff finds that the applicant has correctly applied the input spectra since the spectra envelop the range of soil conditions defined for the AP1000 plant.

For the design of the shield building roof, the applicant used equivalent static analyses with a more refined [ ] finite element model to calculate load demands for the air-inlet region, tension ring, PCS tank wall, and various structural components of the roof. Specifically, the applicant developed a highly detailed linear finite element model of the shield building structure above El. 62.48 m (205 ft). This model, described in Shield Building Report Section 6.2.2, took advantage of the axial symmetry of the shield building above El. 62.48 m (205 ft) to model only a quarter of the building. The applicant used this detailed quarter finite element model because the shield building roof required a more detailed finite element representation to properly capture the demands on each of its structural components. The horizontal input acceleration was an angular acceleration located in the soil beneath the basemat such that the lateral accelerations matched the horizontal accelerations from the SSI analysis. To account for concrete cracking, the stiffness reduction factor of 0.80 times the concrete modulus was utilized in the seismic analysis.

The applicant then combined seismic responses (member forces and deformations) to determine the stresses in some regions of the shield building structure. The Shield Building Report states that the responses of the shield building structure, from the three directions of seismic input, are combined by the square root of the sum of the squares (SRSS) method. However, as clarified in the September 2, 2010 response to RAI-TR85-SEB1-27, and in Shield Building Report, Revision 4, Section 6.2.2, the applicant used the 100-40-40 method for combining the three directions of seismic responses for the shield building roof (tension ring, air-inlet region, W36 beams, conical roof, and PCCWST), the containment, and the basemat. Member forces from the shield building analyses were generated for each element or at critical cross-sections (e.g., the ring girder).

The application of the SSRS method is acceptable to the staff since this method is in accordance with RG 1.92, Revision 2. However, the applicant indicated that use of the 100-40-40 method has reduced the steel reinforcement area by 16 percent when compared to that of the SSRS method (page 3-17 of the Shield Building Report), which the staff believed should not occur when the 100-40-40 method is properly implemented. The applicant addressed this issue for the shield building and the containment in its response to RAI-TR85-SEB1-27 and for the basemat in its response to RAI-TR85-SEB1-32. These two RAIs were addressed and considered resolved. The staff's evaluation of the applicant's response regarding the implementation of the 100-40-40 method is described in Sections 3.8.4.1.1.3.7 and 3.8.5 of the SER.

Design for Concrete Cracking and Steel and Concrete Composite Damping

The applicant stated that its SC wall module is designed in accordance with the strength method in ACI 349. The applicant used a linear elastic analysis finite element computer code, [ ], to quantify the seismic response of member forces in elements for the shield building design. In Section 10.2.1.1 of the Shield Building Report, the applicant stated that for design-basis seismic analysis (Level 1), concrete structures are modeled with linear elastic un-cracked properties with the modulus of elasticity reduced to 80 percent of its value. This reduction is made in order to reduce stiffness and to reflect the observed behavior of concrete when stresses do not result in significant cracking, as recommended in Table 6.5 of FEMA 356.

In Section 3.2.1 of the Shield Building Report, the applicant stated that the SC material damping is assumed to be 5 percent. The staff noted that 5 percent is appropriate for SSE demand and typically invokes a reasonably high response level that includes appreciable concrete cracking. However, the staff was concerned that a reduction factor of 0.8 and 5-percent material damping were incompatible.

In Appendix B to the Shield Building Report, the applicant provided the data on concrete cracking for the shield building (Figures B-18 through B-21) and the auxiliary building (Figures B-48 and B-49) predicted by [ ]. The applicant stated that the predicted concrete cracking for the shield building and auxiliary building was extensive. As a result, the staff could not find the justification for the assumption of a 0.8 reduction factor (for the stiffness ratio) and 5-percent material damping, given the level of cracking indicated in the [ ] analysis. To address this concern, the staff issued RAI-SRP3.7.1-SEB1-19 and requested that the applicant revise its response to RAI-SRP3.8.3-SEB1-03 as appropriate.

In a letter dated July 30, 2010, the applicant updated its responses to RAI-SRP3.7.1-SEB1-19 and RAI-SRP3.8.3-SEB1-03 and provided comparisons of the results of [ ] linear and nonlinear analyses that were time-history analyses based on the envelope of the soil and rock profiles. Comparisons were made at the shield building roof elevation, shield building west wall (at grade elevation), and four other locations in the auxiliary building.

The applicant also provided stress/strain curves for the [ ] linear and nonlinear analyses and showed that cracking was occurring under SSE loading using 5-percent structural damping. The staff reviewed these results and finds the applicant's use of 5-percent structural damping acceptable based on the predictions of seismic demands sufficient to cause concrete cracking.

The staff reviewed the comparisons of ISRS for the analyzed locations and finds only minor differences in response between the [ ] linear and nonlinear models. The small differences in response suggest that the [ ] concrete stiffness reduction factor is a reasonable assumption for SSE loading. However, the applicant did not provide [ ] comparisons for the same locations. Since [ ] is the AP1000 design-basis code, the staff believes that the comparisons of [ ] to [ ] are necessary to validate model similarity.

At the August 18–20, 2010, structural audit, the applicant presented the comparison between the [ ] and [ ] linear analysis results. This comparison sufficiently demonstrated the similarity between the [ ] and [ ] models. In its letter dated September 3, 2010, the applicant updated its response to RAI-SRP3.8.3-SEB1-03 to include the comparisons to [ ].

In conclusion, the staff finds the approach for addressing concrete cracking acceptable. Further, the applicant's studies using [ ], and the correlation of linear results between [ ] and [ ] indicate that a reduced concrete modulus of [ ] and a damping value of 5 percent are justified for the design-basis analysis of the SC wall in the shield building. Therefore, the staff considers these technical issues to be resolved; further discussion appears in Section 3.7.2 (RAI-SRP3.7.1-SEB1-19) and Section 3.8.3 (RAI-SRP3.8.3-SEB1-03) of this report.

In a June 9-11, 2010, meeting, the staff asked the applicant to address concerns about the redistribution of shield building forces resulting from concrete cracking. This item was identified as Action Item 4. To ensure that the dynamic analysis models accounted for the effects of the redistribution of forces caused by shield building concrete cracking, the staff asked the applicant to assess the effects of cracking near the base of the west wall and right above the roof at the auxiliary building. Further, the staff asked the applicant to demonstrate that for SSE-level loading, the maximum in-plane shear stresses remain within the limits allowed by ACI 349.

In its July 30, 2010, letter in response to Action Item 4, the applicant provided the requested comparisons using the [ ] (nonlinear) and [ ] (linear) analysis codes to address the extent of concrete cracking and any needed load redistribution caused by the cracking. The applicant compared concrete shear stress at various locations along the west wall at El. 100′. The results shown in Figures 4-3 through 4-6 of the letter indicate that the in-plane concrete shear stress using [ ] and [ ] remains below 4136 kPa (600 psi) for critical design locations analyzed. The applicant stated that these results demonstrate that the in-plane shear stress is below the allowable shear stress of 4688 kPa (0.85×800 psi = 680 psi) in ACI 349, Section 11.7.5.

The applicant also provided results for in-plane shear distribution at the east wall above the auxiliary building roof. Figure 4-8 provides a comparison of the [ ] and [ ] results and indicates that shear stress is below the ACI 349 allowable limit of 4688 kPa (680 psi).

Based on a review of the applicant's [ ] and [ ] analysis results, the staff finds that the applicant's in-plane concrete shear stresses are below ACI 349 allowable limits at El. 30.4 m (100 ft) and at the east wall above the auxiliary building roof and, thus, finds the results to be acceptable and in accordance with the criteria in NUREG-0800 Section 3.8.4. Therefore, the staff finds the applicant's response to Action Item 4 to be acceptable.

Thermal Loads - Concrete Shrinkage and Thermal Cycling

In both the NRC's letter of October 15, 2009, and Action Items 19 and 20 from the meeting of June 9-11, 2010, the staff raised concerns related to the need for the applicant to consider the effects of concrete shrinkage and thermal cycling loads in the design of the shield building. The staff based its concern, in part, on issues identified in a study by Oliva and Cramer, of the Structures and Materials Test Laboratory at the University of Wisconsin, entitled "Self-Consolidating Concrete: Creep and Shrinkage Characteristics," issued January 2008. The study shows that self-consolidating concrete may exhibit a higher dimension change because of creep and shrinkage than conventional concrete does under shear friction loads. In the Shield Building Report, the applicant predicted extensive vertical cracking because of thermal cycling. As a result, the staff asked the applicant to analyze how the extent of cracking and the load will be redistributed via the design of the shield building to preclude the effects of the cracking on the integrity of the structure.

In response, the applicant reevaluated the thermal shrinkage effect of the in-filled concrete in the SC wall module. After reviewing the parameters used in the thermal shrinkage and thermal cycling analyses, the applicant used a more realistic shrinkage strain value of 200 micrometers per meter (μm/m) ($2x10^{-10}$ inches per inch (in/in)). The applicant stated that the use of the shrinkage strain value indicates that no cracks occurred and the stresses produced on concrete and steel surface plates are extremely low.

The staff believes that the original applicant thermal shrinkage analysis, with the shrinkage strain of 400 μm/m (4×10^{-10} in/in) is conservative because it exceeds the realistic strain value of 200 μm/m (2×10^{-10} in/in). Further, the applicant performed a finite element model analysis using the same three-dimensional finite element model. The finite element model analysis performed was a coupled thermal-mechanical analysis using [ ] 6.9-EF1. This analysis consisted of two approaches—thermal shrinkage and thermal cycling. For thermal shrinkage, an equivalent temperature drop was simulated to produce a uniform thermal contraction in the concrete equal to 200 μm/m (2×10^{-10} in/in). For thermal cycling, a cyclical temperature gradient of 43.33 °C (110 °F) over a 24-hour period was applied. This resulted in a maximum circumferential stress of 2.1 megapascal (MPa) (0.3 ksi) on concrete and -25.8 MPa (-3.74 ksi) on the steel surface plates. The thermal cycling analysis resulted in a maximum circumferential stress of 0.345 MPa (0.05 ksi) on concrete and -2.38 MPa (-2.02 ksi) on the steel surface plates.

The staff reviewed the applicant's reanalysis of thermal cracking and found that the concrete strain of 400 μm/m (4×10^{-10} in/in) is conservative and that vertical cracking is minimal; therefore, the reanalysis is acceptable.

3.8.4.1.1.3.5 Design and Testing for Ductility

In its letter of October 15, 2009, the staff stated that the applicant must demonstrate the adequacy of the design and detailing of the SC wall module to function as a fully composite unit as assumed in the design and analysis. In addition, the staff stated that the applicant must demonstrate that the SC wall module had sufficient ductility to survive earthquakes or tornado winds.

In response to this concern, the applicant made several design changes to the shield building. In the executive summary of the Shield Building Report, the applicant stated that design changes were made to the shield building to improve strength and ductility. These changes included adding [ ] connecting the surface plates to demonstrate that the structure will act as a unit under design-basis events. Further, design changes were made to the SC/RC connection, using mechanical connectors to directly transfer the forces from the SC structure to the RC structure, such that the connection will exhibit strength and ductility during seismic events. The applicant stated that the design of the critical features, such as the SC wall module, the SC/RC connection, and the tension ring/air-inlet region, was verified using benchmarked nonlinear analysis in order to demonstrate the overall strength and ductility of the AP1000 shield building. The applicant further stated that it performed benchmarked analyses (confirmatory analysis) and testing to demonstrate that the design has adequate margin to withstand the SSE in accordance with NRC regulations.

In Section 10.2 of the Shield Building Report, the applicant described the detailed analysis performed to support the basis for estimating the shield building system ductility (or drift ratio). The applicant calculated the drift ratio to access the level of system ductility provided in the shield building. The staff notes that the applicant's definition of drift ratio is the ratio of maximum displacement corresponding to a beyond-design-basis demand (e.g., review-level earthquake and the maximum displacement corresponding to the SSE-level demand. In its June 30, 2010, letter (page 63), the applicant provided an updated comparison of results shown in Table 10.2-5 of the Shield Building Report. The results were obtained using the Level 1-3 analysis models discussed in Section 3.8.4.1.1.3.2 of this SER. The applicant calculated a maximum drift ratio of 6.4 corresponding to the Level 3 analysis displacement (19.6 cm (7.7 in)) from 2.6 SSE loading divided by the SSE-level displacement (3.0 cm (1.2 in)). However, the staff was not able to correlate predicted drift ratios with system ductility. To address this concern, the staff asked the applicant to provide further clarification of its design in relation to ductility. In its response, the applicant supplemented the June 30, 2010, submittal with a letter dated September 3, 2010, which described its philosophy and approach to design and their implications to ductility.

The applicant stated that its design philosophy in relation to ductility is analogous to the "capacity design" approach in FEMA 356-2000, in which the designer identifies a ductile failure mechanism for the overall structure, designates structural fuses that will undergo inelastic deformations and dissipate energy, designs and details the fuses to prevent brittle failure modes from controlling their behavior, and designs the remaining portions of the structure with sufficient strength to resist the force demands delivered by the fuse regions. This approach is referred to as a "strong column-weak" beam design approach in accordance with ACI 349-01, Article 21.4.2.2, for the design of moment-resisting frames.

The applicant's approach is to identify, from the results of the analysis for the calculation of member forces and through confirmatory analysis, the locations in the SC structure that are predicted to become plastic hinges (called fuses by the applicant) when subjected to earthquake forces. In the case of the shield building, this requires earthquake forces beyond the design basis seismic loads. Design detailing for the regions in the shield building assumed to be plastic hinge regions conforms to requirements in ACI 349-01, Articles 21.3.3.1-21.3.3.3, which results in shear reinforcing spacing of depth divided by [ ] maximum. This detailing is intended to prevent brittle failure modes from pre-empting the ductility of the plastic hinge regions. In regions outside of these assumed plastic hinge locations, the applicant's design conforms to Article 21.3.3.4, which requires shear reinforcement ([ ]) spaced at no farther apart than half of the depth dimension. In addition, the design for these regions also provides sufficient strength to meet the calculated design demands. Although the ductility detailing requirements in Sections 21.3 and 21.4 of ACI 349 do not apply to the shield building structure, the applicant invoked them for the analogy of the applicant's design approach to the "capacity design" approach.

Continuing its analogy to the "capacity design" approach, the applicant stated that in regions of high out-of-plane shear demand, close to supports and connections with other structures, [ ]. At the connection to the basemat, this region extends [ ] above the connection region, [ ]. In SC to RC connection regions within the auxiliary building, [ ] spacing extends beyond the connection to about [ ] above and to the side of those regions of the shield building where other structures, such as the shield building roof, attach to the SC wall. The actual distance above this SC to RC connections is, [ ].

In regions away from supports and connections, the AP1000 uses SC modules with [ ], which provides sufficient strength to meet the calculated demands.

The following is the staff's evaluation of the safety of the shield building based on the applicant's method of demonstrating that there is ductility in the design of the shield building.

The staff finds that ACI 349-01, Article 21.4.2.2, is intended for moment frame structures and is not directly applicable to cylindrical shell structures, such as the AP1000 shield building. Cylindrical shells will distribute forces in a manner that differs from a 2D or 3D framed structure. Specifically, cylindrical shells primarily resist seismic lateral loads through membrane action by a combination of in-plane shear, to resist lateral shears, with tensile and compressive forces to resist overturning moments. Furthermore, ACI 349-01 has neither provisions nor requirements for ductility detailing for unique structures, such as the shield building. The staff also finds that the calculation of member forces for the design basis seismic loads for the shield building did not involve load reductions that invoke the formation of plastic hinges for the dissipation of energy. In addition, the applicant's own design methodology for the shield building, based on ACI 349-01, requires that shear strength capacity must be provided everywhere including the assumed hinge locations, which is done for the shield building.

Providing sufficient strength in the plastic hinge regions to meet the calculated shear demands is not a requirement for the "capacity design" approach. For the above reasons, the staff finds that the applicant's design methodology for the design of the shield building to resist seismic loads is not, in a strict sense, a "capacity design" approach.

However, the staff agrees that the inherent premise used in ACI 349, Article 21.4.2.2, of providing ductile detailing where demands are high, can be extended to a cylindrical shell if analysis has been performed to identify locations of high demands, and conservative out-of-plane shear strength to meet the calculated demands is provided elsewhere. For the AP1000 shield building, the applicant provides ductility detailing in the regions of high demands. In the regions of low out-of-plane shear demands, the applicant provides [ ] at a spacing less than one-half of the depth of the wall and conservative demand to capacity ratios

(Reference September 3, 2010 submittal, Figure 4-1, and Reference June 30, 2010, submittal, Figures F1.1.2-1 to F1.2.2-16).

Also in the September 3, 2010, submittal, the applicant stated that cylindrical shells, such as the shield building wall, primarily resist seismic lateral loads through membrane action by a combination of in-plane shear, to resist lateral shear together with tensile and compressive forces to resist overturning moments. Subsequently, the applicant concluded, based on this understanding and the results of a [ ] for seismic loads greater than the design basis loads, an overall ductile failure mechanism would develop in the shield building structure with the structural fuses located in the SC portions of the shield building as designed. According to the applicant, the structural fuses have small inelastic strains and are located either close to the base of the structure, or at support points, or where there are connections to the auxiliary building.

More specifically, also in Section 2.0 of the September 3, 2010, submittal, the applicant states that the [ ] indicates that for seismic loads greater than the design basis loads, the overturning moment and base shear at the base of the structure cause either tension yielding of the steel plates in the SC portion, or tension yielding of the steel reinforcement in the RC portion of the shield building, depending on the loading combination and direction. In this submittal, the applicant also states that for loads greater than the seismic design basis loads, yielding of the steel faceplates from in-plane shear can occur for certain loading directions. Thus, the ductile failure mechanism for the overall structure is governed by the yielding of steel plates or yielding of steel reinforcement in the RC portion of the structure. The applicant then concluded that for loads greater than the design basis loads, the shield building would develop a ductile failure mechanism with structural fuses in the SC portions located as designed.

The staff evaluated the applicant's design approach of providing ductility detailing in the regions of high stresses and of providing the strength necessary to meet the design demands in the regions of low demands and finds it to be reasonable. This approach conforms to the approach in ACI 349-01, Articles 21.3 and 21.4 for moment resisting frames, for which ductility design is required by ACI 349, as opposed to structures such as the shield building structure for which ACI 349 does not have ductility provisions or requirements. The staff also finds that the shield building structure, a complex cylindrical shell, distributes loads in a manner that differs from 2D or 3D frames and can be more uncertain. The staff finds that the shield building design provides conservative demand to capacity ratios in the regions of the wall with [ ] that can account for those uncertainties. Specifically, the calculated demand to capacity ratios for out of plane shear are for the most part less than or equal to 0.3. In addition, the regions of the wall where these demand to capacity ratios are higher than 0.3, and as high as about 0.6 in a few locations, are small in area and localized.

The staff finds that the combination of the low demand to capacity ratios for out-of-plane shears in the regions with [ ] spacing with ductility detailing in the regions of high demands provides reasonable assurance of the building safety under the design basis seismic loads by ensuring that the building has structural capacity in reserve, through a combination of structural strength and ductility, for the seismic design basis loads.

<u>Testing for Strength, Cyclic Loading and Ductility</u>

Section 7.11.1 of the Shield Building Report states that tests were conducted to demonstrate the cyclic behavior and ductility of the SC-portions of the shield building. [

]. Since there are two types of shear loads (the one perpendicular to the wall, which is called out-of-plane shear, and the other along the wall in the hoop direction, which is called in-plane shear) acting concurrently and simultaneously on any point of the shield building during earthquakes, [

]. One type of SC module is used at or near connection regions, which require high shear ductility and strength, and the proposed design and detail for that SC module was to use [ ] between faceplates, and spaced [ ] in both vertical and horizontal (hoop) directions. The other type of the SC modules is used for the remaining portion of the shield building wall with less shear ductility and strength demand, away from the connection regions, and the proposed design and detail for that SC module was to use [ ] between faceplates, [ ] in both vertical and horizontal (hoop) directions. The applicant's acceptance criteria for the ductility tests for each type of module under each kind of shear loads are listed below:

The applicant used the following acceptance criteria for the ductility tests:

<u>Acceptance Criteria for Ductility Tests</u>

For out-of-plane shear, ductility was to be established and measured through a loading protocol as follows:

– [

].

– [

]..

For in-plane shear

– [

].

<u>Out-of-Plane Shear Testing To Demonstrate Ductility</u>

The out-of-plane shear test specimen [ ] tie-bar spacing tested monotonically at shear span a/d=3.5 indicated a brittle failure mode at the load of [ ] and had less strength than the companion specimen tested monotonically at a/d=2.5, which attained a higher load [ ]. The test results for out-of-plane shear showed that the modules with [ ] [ ] failed in a brittle manner and that the case with a [
is the more critical shear case.

However, the staff notes that information provided by the applicant in its supplemental letter dated September 3, 2010, Figure 3-3 indicates that there is sufficient margin between the load corresponding to the maximum SSE-level demand (approximately 80k) and the failure load of the both out-of-plane specimens [ ]. According to the applicant's design methodology this margin will be less than that shown in this figure when only the contribution of the steel is taken into account to account for tensile forces. Even for these conditions, the staff finds that there is significant margin in the specimen to preclude a brittle failure under design-basis (or SSE) loads.

The staff also finds that the tests results show that there is conservatism in the use of the ACI 349 equation for strength, $V_n=V_s+V_c$, for the AP1000 SC structure in that the design strength is bounded by the load at which brittle failure in the SC specimens occurred.

<u>SC Modules under Cyclic Loads</u>

For SC modules under cyclic loads, the applicant stated that the test specimen with [ ] developed its plastic moment capacity and had excellent cyclic behavior during the [ ]. Further, the applicant stated that the specimen demonstrated some strength degradation during the [ ].

The staff reviewed these test data, and concludes that the SC module attained a higher load [ ] than the specimen [ ], and attained a displacement ductility ratio (the displacement value at failure divided by the displacement value at yield) of [ ].

The applicant stated that the specimen with [ ] developed its expected shear strength of [ ] and had excellent cyclic behavior during the [ ]. Some strength degradation during the [ ] cycles was observed, but the shear strength of the specimen was still greater than the expected shear strength. [ ].

The staff reviewed the test data, and finds that the applicant defined the yield displacement at the point at which the specimen achieved the strength (V_c + V_s), which is different from the Δy definition of Δy as stated for the above module with [ ], and is incorrect for this test. By judging the hysteretic curves, this test specimen had not been loaded to sufficiently high displacements to induce yielding of the steel faceplates. Therefore, referring to the loading cycles as [ ], as stated by the applicant, is incorrect. The applicant addressed the staff's concern by removing [ ] signs from the figure in its September 3, 2010, submittal.

However, the applicant provided in the September 3, 2010 submittal on ductility, Figure 4-2, which shows the measured cyclic shear force mid-span displacement response of the specimen [ ]. The staff finds that the cyclic test response shows [ ]. Further, the out-of-plane shear strength of the non-fuse specimen under cyclic loading can still be estimated using the ACI 349 Code equations and the specimen exhibited adequate cyclic load behavior at load levels equivalent to calculated out-of-plane shear demands.

The staff finds that testing of SC wall modules with [ ] spacing did not demonstrate that the SC wall module is ductile because it did not meet acceptance criteria for ductility as proposed by the applicant.

Nonetheless, in the staff's view, the SC module [ ], although it failed in the first cycle at [ ], showed appreciable ductility and is expected, if it were tested at [ ], to result in reasonable ductility in the design. Therefore, in the staff's view, this test demonstrates that sufficient ductility capacity exists for the SC module [ ].

<u>In-Plane Shear Cyclic Testing To Demonstrate Ductility</u>

In Section 7.12 of the Shield Building Report, the applicant described the in-plane cyclic shear tests designed to demonstrate the cyclic behavior and ductility of the SC shield building design for in-plane shear loading. [ ].

The staff's review of the test plan for the in-plane shear test (Section 7.12) finds that the test model and test set-up boundary conditions [ ], as shown in Figures 7.12-1 to 7.12-5, may provide additional resistance and can lead to an over-estimation of the actual strength of the SC wall module. The applicant had to terminate the test after [ ] due to laboratory safety constraints and, therefore, could not complete the ductility test. The staff believes that cyclic loading beyond the yield point is needed to ascertain the ductility of the SC module and to observe the deterioration of the concrete between the faceplates.

In the September 3, 2010, submittal, the applicant provided a plot of the [ ] (Figure 5-2) and an envelope plot of cyclic lateral load (Figure 5-3). The applicant stated that the test results demonstrated that the SC specimen could undergo loads with acceptable deformations up to [ ] the SSE level.

The staff's review finds that the test was inconclusive with respect to demonstrating ductility. However, the applicant, in Section 5.1 of the submittal dated September 3, 2010, described tests on SC modules conducted by Ozaki et al. (2004) to supplement the basis for demonstrating ductile in-plane behavior. These tests on SC panels were performed to determine the cyclic in-plane shear and to evaluate the effects of various plate parameters, such as plate thickness and axial force. One of the test specimens, S4-00NN, was judged by the applicant to be the most relevant to the AP1000 SC module. [ ]. The ratio of shear stud spacing to plate thickness is 30 for specimen S4-00NN and 11.33 for the AP1000 SC module. Consequently, the applicant concluded that the behavior of the AP1000 SC module will be slightly better than that of the S4-00NN specimen. Specimen S4-00NN had a measured ductility value, defined as ultimate strain to yield strain, of 2.82, as shown in Figure 5-1 of the September 3, 2010, submittal.

The staff reviewed the Ozaki paper, and found that the test was properly conducted and credible. In SER Table 3.8-1, staff performed a review of the Ozaki, et al. paper to compare a few key parameters of the AP1000 design and the S4-00NN specimen. Based on this comparison, and the good agreement of SC parameters, the staff finds the applicant's use of the test data to demonstrate ductility of the SC wall to be appropriate.

Table 3.8-1. Comparison of Test Specimen of S4-00NN and AP1000 SC Module

Parameter	Test Specimen S4-00NN	AP1000 SC Module
SC wall thickness/faceplate thickness	44.4	[ ]
Stud spacing/wall thickness	0.67	[ ]
Stud spacing/plate thickness	30	[ ]
Concrete compressive strength MPa (psi)	42.79 (6,206)	[ ]
Steel plate yield stress MPa (ksi)	346.1 (50.2)	[ ]

The staff finds that although there were concerns regarding the test setup at Purdue, the test results indicate that the design for the in-plane shear strength criteria used ([ ]) is adequate.

In addition, the staff finds that although the Purdue test specimen was actually a framed shear wall and the stiffness of the frames was added to that of the wall during the test, the test results (reported in the Osaki paper) help assure the staff of the behavior of the SC wall module under SSE loads.

<u>Conclusion of Design and Testing Related to Ductility and Safety of the Design</u>

In summary, the staff finds that the purpose of shear tests is to establish the minimum shear reinforcement ([ ]) to the SC module so that it can function as a unit to resist both out-of-plane and in-plane shear forces, provide sufficient ductility (energy absorption/dissipation capability) for seismic-induced energy, and provide sufficient stiffness for the shield building to meet the allowable building drift limit. The staff finds that the tests were an acceptable basis to establish this minimum.

The staff finds that ACI 349 (Article 21.4.2.2) is intended for moment frame structures and is not directly applicable to cylindrical shell structures, such as the AP1000 shield building. Cylindrical shells distribute forces in a manner that differs from a two- or three-dimensional framed structure. However, the staff agrees that the inherent premise used in Article 21.4.2.2 (providing ductile detailing where demands are high) can be extended to a cylindrical shell if analysis has been performed to identify the locations of high demands.

Also, the staff finds that for the AP1000 shield building, the applicant provided ductility detailing in the regions of high demands. In the regions of low out-of-plane shear demands, the applicant provided conservative demand-to-capacity ratios (Figure 4-1 of its June 30, 2010, submittal and Figures L.4-23 and L.4-24 of the Shield Building Report, Revision 4). The staff finds this approach to be acceptable.

In addition, the staff finds that the AP1000 shield building design has [ ] spacing to ensure that the SC modules will function as a unit. For the regions of the SC wall with higher out-of-plane shear loads, and where yielding of the SC wall would be expected to initiate under a combination of tensile forces and out-of-plane bending for seismic loads in excess of the design-basis loads, the applicant detailed the SC modules with [ ] spacing to provide out-of-plane shear ductility. For the regions of the SC wall with low out-of-plane shear demands [ ], and the SC wall detailing does not provide out-of-plane shear ductility. In these regions, the out-of-plane shear demands calculated by the applicant are low and the SC wall modules as detailed provide conservative strength demand to capacity ratios.

For the in-plane shear test, the staff finds that the test results indicate that the design for the ACI 349 the in-plane shear strength criteria used, [ ] is adequate. The test results were inconclusive with respect to measurable ductility. However, cyclic ductility tests performed in Japan (documented in the Ozaki paper) indicate that the wall will exhibit ductile behavior under cyclic in-plane shear. On these bases, the staff concludes that the SC wall will provide adequate strength, stiffness, and ductility under design-basis (or SSE) seismic loads.

The staff finds the design for strength, stiffness, and ductility to be acceptable.

3.8.4.1.1.3.6 Design of the Steel and Concrete Composite-to-Reinforced Concrete and Basemat Connections

Section 4.1.1 of the Shield Building Report describes the design details for the revised shield building connection. The applicant stated that the steel liner plates are connected to the RC wall reinforcing bars by [ ] of the SC/RC connection (Figures 4.1-2 through 4.1-5 of the Shield Building Report). [ ]. The [ ] connection is designed to the allowable working stress limits of ANSI/AISC N690 for loads in the reinforcing bars equivalent to 125 percent of the yield strength of the specimen.

In its review of the SC/RC connection design, the staff identified several concerns discussed at the June 9-11, 2010, meeting and documented as action items. In Action Item 7, the staff asked the applicant to clarify the design and load path for the SC/RC connection. In Action Item 8, the staff asked the applicant to provide justification that voids in the SC/RC connection region would not affect the load path in compression. In Action Item 9, the staff asked the applicant to provide verification that calculated shear friction values in the SC/RC connection are below the ACI 349 allowable limit. In Action Item 11, the staff asked the applicant to identify the type of [ ] connector used for the shield building, in accordance with ACI 318, Chapter 21, and to justify the use of [ ], as appropriate.

In its August 3, 2010, letter, the applicant provided responses to the above action items. In response to Action Item 7, the applicant, in Table 2.1.1-1, stated that it would implement a design change to the SC/RC connection. The applicant stated that [ ] will be used to connect the #14 reinforcement bars in the basemat to the [ ] connection. In addition the applicant compared connection yield capacities of the SC/RC connection components, such as the [ ]. In addition, the applicant summarized the stress ratio (i.e., demand to capacity ratio) for the various loading conditions on the SC/RC connection components. In Table 2.1.1-2, the applicant provided the stress ratios for tension (0.37), compression (0.84), moment (0.08), in-plane shear (0.84), out-of-plane shear (0.05), and combined tension, bending, and in-plane shear (0.64).

Based on the applicant's description and data for the SC/RC design change, component capacities, and component stress ratios that are all less than one, the staff finds the applicant's response acceptable.

Further, for Action Item 7, the applicant described the load path and showed that with the combination of 2.5 cm (1 in) thick liner plate, 5.1 cm (2 in) support plate, 5.1 cm (2 in) gusset plate, [ ], the RC/SC connection can transfer loads from tension, compression, bending moments, and shear. Hence, the load path is established through the SC/RC connection and is acceptable to the staff.

In response to Action Item 8, the applicant stated that small gaps under the connection support plates will not affect the load transfer in compression. The applicant stated that the gap under the support plates is considered for the calculation of the capacity of the connection for compression forces, as shown in Figure 2.1.2-1. Further, the direct transfer of compression force through the concrete is only considered in the region between the support plates. The applicant calculated a compression ratio for the concrete between the support plates to be less than one (0.84). Based on the applicant's calculation of compression ratio, which neglects the concrete contribution beneath the support plates, the staff finds the response to Action Item 8 to be acceptable.

In response to Action Item 9, the applicant stated that since the design of the SC/RC connection was changed from smooth bars to deformed reinforcement bars, the ACI 349 Code was applicable. The applicant calculated the SC/RC shear capacity in response to Action Item 7 and provided the demand-to-capacity ratios in Table 2.1.1-2. The reported demand-to-capacity ratio for in-plane shear was 0.84 and for out-of-plane shear was 0.05. This indicates that the capacity of the connection is 16 percent higher than the demand. Based on the applicant's design change from smooth to deformed reinforcement bars and the shear capacity being within ACI 349 limits, the staff finds the response to Action Item 9 to be acceptable.

In response to Action Item 11, the applicant stated that it will use the ACI 318 Type 2 mechanical splice and revised its qualification and production criteria for the Type 2 connectors in compliance with the ASME B&PV Code, Section III, Division 2, Subsection CC, "Code for Concrete Containments," Article CC-4333. In addition, the applicant will use the reinforcement mechanical splice examination criteria as defined by Article CC-5320. Based on this change, the staff finds the response to Action Item 11 to be acceptable.

Based on the applicant's responses to the above action items, the staff considers the design of the SC/RC connection to be acceptable. The staff notes that the applicant will provide a COL information item that will address the constructability of the shield building, including the SC/RC connection. Section 3.8.6 of this SER discusses and evaluates this COL information item.

<u>Testing of the Steel and Concrete Composite-to-Reinforced Concrete Connections</u>

In Section 7.3 of the Shield Building Report, the applicant stated that a full-scale anchorage test was performed to demonstrate the strength and ductility of the previous SC/RC connection design and its ability to develop the steel reinforcement on either side of the connection. Although the test specimen was representative of an earlier connection design, the applicant stated that the test specimen had some similarities with the revised connection. The test was also used to benchmark the [ ] analysis code for use in detailed analysis (Section 8.10 of the Shield Building Report).

In Section 7.13 of the Shield Building Report, the applicant described the results of the anchorage tests and found that the objectives and acceptance criteria were satisfied. The test demonstrated the capability of the SC/RC connection to transfer 125 percent of strength of the [ ] and the ductility of the connection region.

The staff's review of the test results confirmed that the SC/RC connection exhibited adequate strength and ductility to transfer 125 percent of the strength of the [ ]. Although the test was representative of the previous design, the staff considers the new design to have improved capacity because the [ ] bar connects [ ] to the support and liner plates. As a result, the staff does not believe that further testing is required for the SC/RC connection.

The staff finds the applicant's design of the SC/RC connection acceptable based on the applicant's revised design, demonstration of design stresses below code-allowable limits, the use of a [ ] mechanical [ ], and the anchorage test that involved testing of a connection with some similarities to the current design of the connection.

3.8.4.1.1.3.7 Design of the Tension Ring and Air-Inlet Region

Chapter 5 of the Shield Building Report describes the design of the tension ring and air-inlet structure. The tension ring is located at the interface of the SC air-inlet structures and the shield building RC roof (Figure 5.1-2 of the Shield Building Report). The top of the tension ring interfaces with the RC roof slab. The tension ring supports [ ] steel roof girders that are located under the RC roof slab. The bottom of the tension ring is attached to the air-inlet structure. The bottom of the air-inlet structure is attached to the top of the cylindrical SC wall of the shield building. The applicant revised the design of the tension ring in the Shield Building Report and reduced the air-inlet areas to provide more concrete for structural strength to the air-inlet region. The steel box girder for the tension ring consists of two closed sections, both of which are filled with concrete. The top section is triangular in cross-section and has sloping top surfaces in order to interface with the RC roof slab. The bottom section is rectangular in cross-section, with steel flanges and webs.

The air-inlet structure is an SC structure [ ]. The top of the faceplates of the air-inlet structure [ ]. The steel faceplates are connected together by [ ] vertical spacing. The air-inlet structure is an SC structure with through-wall openings for air flow. The air-inlet pipes are connected to the infill concrete by welded shear studs on their outside surface. The air-inlet openings consist of [ ]. The air-inlet pipes, spaced at approximately [ ] is poured into the air-inlet structure between the faceplates and bonds to the [ ] of the faceplates and the [ ] of the air-inlet pipes. That bonding makes the air-inlet SC structure act as a unit. The [ ] thick steel plates on each face, aligned with the inner and outer flanges of the tension ring, serve as primary reinforcement. The concrete infill is connected to these steel plates with [ ]. The steel face plates at the top of the air-inlet structure [ ] on the underside of the bottom tension ring web plate also function to attach the tension ring to the air-inlet structure. The faceplates at the bottom of the air inlets structure are welded to the faceplates of the SC wall.

The staff finds that the applicant's changes in the design of the tension ring girder, from an [ ], have resulted in a much improved design primarily because the design change makes the tension ring girder consistent with proven methods in ANSI/AISC N690. This change also provides a more predictable load path and stiffens the tension ring structure.

The tension ring is designed as a [ ], according to the design of the member forces in ANSI/AISC N690, and the concrete infill is credited only for stability of the steel plates. The design loads for the tension ring and air-inlet structure are established from the [ ] linear analysis. The tension ring is designed to have high stiffness and to remain elastic under required load combinations. The air-inlet structure was designed as an SC module.

In Section 5.1 of the Shield Building Report, the applicant stated that the current plan for construction of the air-inlet structure and tension ring is for the structures to be [ ] below the bottom of the tension ring.

As a result of its review, the staff raised a concern with the applicant (Action Item 13) that a construction joint in the air-inlet region [ ] below the tension ring would reduce the shear capacity of the concrete in this critical section. During construction, [ ] is poured through the holes in the horizontal web plate, and it is expected that the [ ] would flow and fill up to the top of the construction joint. The staff questioned whether the construction method for the tension ring girder/air-inlet region would disrupt the integrity of the structure and whether it would function as designed under design-basis loads.

In its June 30, 2010, letter response (page 93), the applicant provided a calculation to address shear friction loads at the air-inlet connection and construction joint in the tension ring. The applicant calculated the shear capacity of the air-inlet connection (based on ACI 349) to be [ ]. As a result, the applicant concluded that the capacity of the construction joint is governed by the shear transfer at the plate at the bottom of the ring girder-to-wall interface and not by shear transfer at the plane at the construction joint. The applicant also stated that this construction joint will be prepared by intentional roughening, in accordance with the requirements of ACI 349, Article 11.7.9.

The applicant also performed a calculation for the capacity of the shear ties to show that they are adequate to address the tapered transition from the [ ] thick SC wall to a [ ] thick air-inlet wall (page 96 of the June 30, 2010, letter). The calculation assumed an axial force demand of [ ] coupled with [ ] acting in tension (lower end of the taper) and [ ] acting in compression (upper end of the taper). The applicant assumed that over a height of 0.61 m (2 ft), the [ ] have a capacity [ ]. At the elevation of the transition, the maximum out-of-plane shear [ ]. As a result, the applicant stated that the [ ] can be credited for both tension caused by the inner plate transition and the out-of-plane shear demand. At the top of the transition, the applicant calculated a maximum compressive force [ ], resulting in a [ ] demand [ ].

The staff reviewed the results of these calculations and finds that the calculations' assumptions and technical bases are based on ANSI/AISC N690 and the criteria in ACI 349, and are, therefore, acceptable for the design of the tension ring and air-inlet region of the shield building. However, the staff notes that in the June 30, 2010, letter response (page 96); the applicant stated that because of the amount of congestion in this area, constructability studies are being performed. These studies will evaluate whether the current tie-bar configuration is adequate for concrete placement and will provide insight into design details that would enhance the design. During final design detailing, the applicant will consider increasing tie-bar capacity in this region based upon the results of the constructability studies.

As discussed in "Determination of Responses to Earthquake Loads" in Section 3.8.4.1.1.3.4 of this SER, the applicant did not properly implement the 100-40-40 combination method for seismic loading from the three earthquake directions (x, y, and z) when designing the tension ring and air-inlet regions. The applicant addressed this issue in its response to RAI-TR85-SEB1-27. Section 3.8 of the SER for the AP1000 DCD describes the staff's evaluation of the applicant's response about the implementation of the 100-40-40 method. The applicant's draft response to RAI-TR85-SEB1-27, transmitted on September 23, 2010, provided tabulations for the air-inlet region and tension ring to demonstrate the adequacy of the design using the applicant's 100-40-40 method. The staff's review of these tabulations determined that the applicant's 100-40-40 method results in lower member demands than the SRSS approach (the accepted method in RG 1.92). However, there were still substantial margins when the required member demands using the SRSS combination method were compared to the provided reinforcement for the air-inlet region and to the stress allowable values for the tension ring.

Based on the staff's review of the applicant's detailed design and analysis of the tension ring and air-inlet region as discussed above, the staff finds the design of the tension ring and air-inlet region to be acceptable.

The staff also performed a review of the seismic analysis of the PCS tank described in Section 6.2.2 of the Shield Building Report, Revision 4. The staff's review of this section found that the methodology used for the seismic analysis was consistent with AP1000 DCD Section 3.7.2 except for the use of the 100-40-40 method for all of the roof structural elements. Therefore, during the June 20-24, 2011 audit, the staff requested that the applicant justify the implementation of its 100-40-40 method. In Item 9 of the June 27, 2011 letter submittal the applicant performed additional analyses of the PCS tank using the SRSS method and compared it to the Westinghouse 100-40-40 method. The results showed that the calculated steel areas required tp meet design loads using the SRSS method were greater at some locations. However, the staff's review found that even with the higher SRSS results, the calculated steel areas required were still less than the steel areas provided in the design of the PCS tank.

Based on the staff's review of the applicant's detailed design and analysis of the PCS tank discussed above, the staff finds the design of the PCS tank to be acceptable.

3.8.4.1.1.3.8 Design of Roof and Tank Support

The cylindrical section of the shield building structurally supports the roof, which includes the PCCWST. The PCCWST has a stainless steel liner that provides a leak-tight barrier on the inside surfaces of the tank. The shield building PCCWST and the shield building roof are designed as RC sections in accordance with ACI 349. One of the significant loads on the PCCWST roof, and supporting shield building walls, is the seismic loading. To determine the seismic loading on the PCCWST, specific procedures need to be considered. The Shield Building Report indicates that the analysis and design took into account hydrodynamic loads (caused by sloshing during a seismic event) on the PCCWST walls. Detailed calculations were performed in accordance with the procedure described in ASCE 4-98. The finite element model considered the seismic loading of the water, which consists of the impulsive mode (effective fluid weight that acts as a rigid mass) and the convective mode (effective fluid weight that represents the sloshing mass).

Since the mass of water at the top of the shield building is significant, and to ensure that the seismic hydrodynamic loading of the water was properly considered in the analysis and design of the PCCWST and the shield building structural supporting members, the staff asked the applicant to describe in greater detail its method for calculating the seismic loading. Action Item 21 in the June 30, 2010, submittal asked the applicant to describe: (1) how it determined the seismically-induced pressure distributions of the water in the tank; (2) the maximum sloshing height of the water surface; (3) how it considered the potential sloshing impact forces on the tank roof; and (4) how it determined the maximum deflections of the supporting beams to the shield building roof and tank in order to demonstrate that these deflections meet code deflection limits.

In the RAI response, dated September 3, 2010, the applicant provided information to address the seismic-induced pressure distributions, sloshing height, and deflections of the supporting beams to the shield building roof and tank. Based on the staff's independent calculation, the staff found acceptable: (1) the magnitude of the hydrodynamic pressure at the bottom of the outer tank wall used to determine the hoop stress in the tank wall; (2) the hydrodynamic base shear used to calculate the shear stress in the tank wall; (3) the hydrodynamic moment on a section immediately above the tank base used to calculate the axial stress in the tank wall; (4) the hydrodynamic moment on a section immediately below the tank base used to design the tank supporting structure; and (5) the calculation of the water sloshing height used to ensure the water does not impact the tank roof. In addition, the maximum deflection of the supporting radial beams was within code limits. As a result, Action Item 21 is resolved and the design of the PCCWST is acceptable to the staff.

3.8.4.1.1.3.9 Use of Self-Consolidating Concrete

One of the staff's key issues, as identified in its October 15, 2009, letter, was that the applicant consider the self-consolidating concrete material properties and their effects (i.e., higher shrinkage and creep strains, less shear resistance and ductility) when compared to those of standard concrete. In its response, the applicant stated that in the Shield Building Report the use of self-consolidating concrete in the shield building would be limited to selected regions of the structure, including the knuckle regions of the roof, the tension ring, the air inlets, and selected portions of the SC-to-RC connection. Other portions of the structure would be constructed of standard concrete. Both the standard concrete and the self-consolidating concrete would have a compressive strength of f'c = 41.37 MPa (6,000 psi). The applicant stated that standard concrete will be used in most parts of SC construction, with limited use in a few congested areas. The applicant addressed concrete placement, shrinkage, and creep characteristics of the concrete and their effects on the shield building design.

The predicted compressive stress in the steel plate from concrete shrinkage would be 62.05 MPa (9,000 psi), and the stress in the concrete would be 387 psi. The concrete stress is slightly higher than $4\sqrt{f'c}$ = 2.14 MPa (310 psi). However, this is a very conservative estimate because the elastic modulus is lower and there is significant tensile creep at early ages when the shrinkage rate is largest. During the meeting on June 9-11, 2010, the staff asked the applicant, in Action Item 10, to further clarify the use of [ ] and the specific locations where it will be used for the shield building. In response to the action item, in its letter of June 30, 2010, the applicant stated that [ ] is used in select locations in the enhanced shield building where access is limited for a vibrator. The applicant also specified that [ ] is to be placed in the air inlets from about El. 75.0 m (246 ft) up to the top of the tension ring to about El. 83.8 m (275 ft), and below the PCCWST from about El. 89.6 m (294 ft) to about 94.2 m (309 ft).

Based on the applicant's explanations and evaluations regarding the specific concrete strength, its properties, the considerations for limiting the placement of the [ ] only to the congested areas, and the limited use of the [ ] throughout construction of the shield building to help enhance the integrity of the structure, namely in the air inlet regions and below the PCCWST tank, the staff finds the applicant's use of [ ] to be acceptable.

3.8.4.1.1.3.10 Daily Temperature and Thermal Effects

In its October 15, 2009, letter, the staff identified an issue that the applicant had not formally addressed: the daily and seasonal thermal cycling effect on the SC modular construction. In order to address the thermal cycling effect, the applicant performed thermal analysis to quantify the effect of daily and seasonal thermal cycling on the cylindrical wall.

The applicant used a cyclical temperature gradient of magnitude 43.33 °C (110 °F) over the course of 1 day to evaluate the effects of thermal cycling on the SC wall. The assumed temperature cycle is applied to the exterior shield building environment while maintaining an interior building temperature of 21.11 °C (70 °F). The result of the analysis indicated that the maximum stress in the wall is circumferential tensile stress of [ ], which is below the fatigue limit. The applicant concluded that the daily temperature cycling would not cause a fatigue problem. Based on its review of the applicant's analysis, the staff finds the applicant's evaluation of daily temperature and thermal effects acceptable.

3.8.4.1.1.3.11 Combined Normal Operating Thermal and Seismic Demands

During the review of Shield Building Report APP-1200-S3R-003, Revision 3, the staff identified that the applicant had not provided information relating to the combination of normal operating (i.e., ambient) thermal and seismic demands as required by ACI 349, Chapter 9, code provisions. The staff held public meetings on April 12, 2011 and May 17, 2011 to discuss the applicant's plan to address the issue. During the May 17 meeting, the applicant committed to revise APP-1200-S3R-003 to include the load combination of thermal and seismic demands. On June 13, 2011, the applicant submitted Shield Building Report APP-1200S3R-003, Revision 4, which included Appendix L, "Combination of Seismic and Thermal Loads." Shield Building Report, Appendix L, describes; (1) the three-dimensional steady state heat transfer analysis performed to develop thermal demands, (2) the development of reinforced concrete stiffness reduction ratios, and (3) the results of the combined thermal and seismic demands.

Heat Transfer and Thermal Stress Analysis

In Shield Building Report, Section L.1.1, the applicant assumed that the air-inlet, tension ring, and RC conical roof have no significant thermal demand from ambient thermal conditions due to negligible temperature differences between the inside and outside surfaces of these elements. The applicant identified the key regions of the shield building that see significant thermal demand as the region of the cylindrical wall adjacent to the annulus seal (approximately 40.7 m (133.5 ft) elevation), the connection of the cylindrical wall boundaries to the auxiliary building, and the outer wall of the passive containment cooling water system (PCS) tank. For the analysis of the shield building cylindrical wall and its connection to the auxiliary building, the applicant performed detailed three dimensional steady state heat transfer analysis, using the ANSYS NI05 shell model. This model accounted for thermal conduction and convection on the nuclear island building surfaces to provide a more realistic assessment of thermal gradients in key areas. A summary of temperatures used for the analysis is presented in Table 3.8-2 below. The applicant stated that the reference temperature for all materials was 21.1 °C (70 °F) and that the winter temperature condition (rows 2 and 4) was the controlling case for design.

Table 3.8-2, Assumed Temperatures in Shield Building Heat Transfer Analysis

	Location	Assumed Temperature °C (°F)
1	Below Grade	21.1 (70)
2	External Air (Winter)	-40 (-40)
3	External Air (Summer)	46.1 (115)
4	Upper Annulus Air (Winter)	-40 (-40)
5	Upper Annulus Air (Summer)	46.1 (115)
6	Internal Ambient	21.1 (70)

For the calculation of convective heat transfer coefficients, the applicant partitioned the exposed nuclear island building surfaces into three simplified geometries, a cylinder for the shield building exterior wall, a concentric annulus for the shield building upper annulus, and flat plate geometry for the auxiliary building roof and walls. The associated types of forced convection equations for these geometries were a cylinder in a cross flow, flow in a tube with concentric annulus, and mixed flow over a flat plate.

The assumed air (wind) velocity on the exterior walls was 6.25 m/s (20.5 ft/s), and the air velocity inside the upper annulus was 2.1 m/s (7.0 ft/s). The exterior air velocity is representative of weather data over 44 years from the Duluth, Minnesota Airport, which relates peak wind velocity to air temperature. The staff's review of the meteorological data from Duluth, Minnesota is discussed in Section 3.8.2.5 of this report. The applicant then conservatively assumed that the duration of the peak air velocity was sufficiently long (i.e., many hours) to achieve a steady state condition. The values of calculated forced convection coefficients ranged from 8.0 to 15.3 W/m^2 °C (1.4 to 2.7 BTU/(hr-ft^2-°F)) and were applied to each exposed surface.

Results from the NI05 heat transfer model (i.e., temperatures on inside and outside wall faces) were applied to a separate thermal stress analysis model using the NI05 shell element geometry to obtain thermal member forces. To account for concrete cracking from thermal stresses, the applicant reduced the calculated member forces and moments from the finite element model by multiplying them by assumed stiffness ratio factors (cracked/uncracked concrete modulus ratio). Seismic analysis of the AP1000 nuclear island (reviewed in

Section 3.7 of this SER) was performed assuming a 0.8 stiffness reduction factor to account for concrete cracking under seismic demands. For thermal analysis of SC modules and RC structures, the applicant assumed a stiffness reduction factor of 0.625, which results in an effective cracked-to-uncracked stiffness ratio of 0.5 (or 0.8 x 0.625). For concurrent axial tension and flexure, the axial stiffness was reduced to that of the steel plates for a resulting stiffness ratio of 0.22. The ratio of 0.22, to account for direct tension, was used in areas where maximum principal stresses exceeded the concrete cracking threshold represented by the direct tension capacity $[4(f'_c)^{0.5}]$ as defined in ACI 209R, "Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures," Equation 2-4.

Shield building thermal demands were combined with seismic demands. Seismic demands were calculated using the NI05 ANSYS response spectrum model. The NI05 seismic model is described in AP1000 DCD Section 3G.2.2.4. The staff's review of the NI05 model is described in Section 3.7.2.4 of this report.

PCS Tank Analysis

Shield Building Report, Section L.4.4 describes the seismic and thermal load combination for the PCS tank. For the PCS tank, the applicant performed representative one-dimensional heat transfer analysis and analyzed for winter and summer conditions (see Table 3.8-2 above). The analysis assumed the same heat transfer coefficients as used for the SB cylindrical wall. The results of this analysis, which were distributions of inside and outside surface temperatures, were used as input to a more detailed ANSYS quarter-model (reference Shield Building Report, Figure 10.2-10) that used solid elements for concrete material. This model calculated nodal temperatures within the concrete wall, which were applied to a detailed quarter-model to determine thermal stress demands. Seismic demands, including PCS tank hydrodynamic loads from sloshing, were combined with thermal demands. The seismic analysis of the PCS tank is described in Shield Building Report Sections 6.2.2 and H.4.3.2 (hydrodynamic loads). The staff's review of the seismic analysis is described in Sections 3.8.4.1.1.3.4 and 3.8.4.1.1.3.8 of this report.

Results

Shield Building Report, Section L.4.1.1 provides results for the combination of thermal and seismic demands for the SB cylindrical SC wall. For the SC cylindrical wall region, plots (Figures L.4-17 through L.4-22) are provided showing the required SC plate reinforcement for the vertical, circumferential (hoop), and out-of-plane shear directions. These plots indicate that the provided SC plate reinforcement exceeds the ACI 349 code required reinforcement. The minimum ratios of provided, versus code required plate reinforcement, were 1.12 (vertical), 1.06 (hoop), 1.50 (out-of-plane shear; Type II Module tie-bar spacing), and 1.53 (out-of-plane shear; Type I Module tie-bar spacing). Table L.4-2 provides summary results for member forces with maximum out-of-plane shear demand. The reported maximum demand-to-capacity ratios for the wall modules are 0.648 (Type I Module tie-bar spacing) and 0.651 (Type II Module tie-bar spacing).

For the PCS tank, Shield Building Report, Section L.4.4, provides a summary (Tables L.4-4 through L.4-6) of required versus provided reinforcement ratios for the vertical, circumferential (hoop), and out-of-plane shear directions. These tables indicate that the provided steel reinforcement for the PCS tank exceeds the ACI 349 code required reinforcement with margin. The minimum ratios of provided, versus code required reinforcement, were 1.25 (vertical), 1.09 (hoop), and 6.70 (out-of-plane shear).

<u>Evaluation</u>

The staff performed a review of the applicant's technical basis for the seismic and thermal load combination for the shield building cylindrical wall and PCS tank as described in APP-1200-S3R003, Appendix L. The staff's review of Appendix L finds that for the analysis of the AP1000 shield building, thermal forces and moments were reduced in accordance with ACI 349-01 provisions. The staff notes that ACI 349-01, Appendix A, "Thermal Considerations," states that the thermal evaluation may be based on cracked section properties to account for redistribution of internal forces and strains due to concrete cracking. The use of cracked concrete section properties results in a reduced section modulus and subsequent reduction of thermal forces and moments. The applicant assumed reduction factors that were comparable to those recommended in FEMA 356, Table 6-5,"Effective Stiffness Values," and ASCE/SEI 43-05, Table 3-1, "Effective Stiffness of Reinforced Concrete Members. Based on the (a) commitment to use ACI 349 for the design of the shield building, which allows for reductions in thermal forces and moments, and (b) the use of concrete stiffness reduction factors comparable to those recommended in FEMA 356 and ASCE/SEI 43-05 standards, the staff finds the applicant's method of accounting for concrete cracking from thermal stresses to be acceptable.

On June 20-24, 2011, the staff performed an audit of significant Westinghouse calculation reports that support results described in Shield Building Report, Appendix L. These reports related to the shield building cylinder heat transfer analysis (APP-1200-S2C-126), PCS tank heat transfer analysis (APP-PCS-M3C-028), and analysis and design of the PCS tank (APP-1278-CCC-007). In a June 27, 2011 letter, the applicant provided responses to staff audit questions.

Based on the applicant's acceptable method for accounting for concrete cracking and the seismic analysis of the SB cylindrical wall and PCS tank performed in accordance with NUREG-0800 Section 3.7, the staff finds the applicant's consideration of the thermal and seismic load combination for the shield building satisfies ACI 349 code provisions and is therefore acceptable.

3.8.4.1.1.3.12 Local Buckling Analysis

During its review of Revision 1 of APP-1200-S3R-003, the staff found that the applicant had not provided sufficient information to demonstrate that the SC design addressed the effects of local buckling of the SC module faceplates. In response to the staff concerns, the applicant revised the design of the SC wall module by increasing both the inner and outer plate thickness from [ ]. In Section 3.3.1 of the Shield Building Report, the applicant summarized the adequacy of surface plates to resist buckling.

The applicant assumed that the buckling modes for analysis were horizontal ripples caused by vertical loading [ ], vertical ripples caused by horizontal loading [ ], and diagonal ripples caused by in-plane shear loading [ ]. Based on these wavelengths, the applicant concluded that the longest wavelength [ ] controlled the design. The applicant assumed the plate to behave as a [ ] long column, with partial moment restraint at the ends. Appendix A to the Shield Building Report provides the empirical relationships used to evaluate the SC plate buckling capacity. The applicant referenced testing conducted to support the finding that [

]. This buckling stress is lower than the Euler value. Using these assumptions, the applicant calculated the elastic buckling stress of [ ]. Since this buckling stress exceeds the steel plate yield stress, the applicant concluded that inelastic properties of the plate govern.

The applicant verified the performance of the steel plate under construction loads and found that the midspan deflection between [ ]. This deflection resulted in a maximum steel stress of 19.3 MPa (2.8 ksi). As a result of these small displacements and stresses, the applicant concluded that the effect of wet concrete loads on reducing buckling capacity was minimal.

In Section 3.3.1 of the Shield Building Report, the applicant stated that the compression loads in the shield building cylindrical wall are well below the strength of the section. The maximum compression is [ ].

The staff reviewed the applicant’s technical basis for analyzing steel plate buckling, including empirical buckling relationships, in Appendix A to the Shield Building Report and finds the basis acceptable given the geometric similarity of the tested panels with the AP1000 design. On the basis that the applicant has performed a buckling analysis using acceptable empirical design equations and that the applicant has predicted relatively low compressive stresses from all load combinations, the staff finds the applicant’s design to resist local buckling of steel plates to be acceptable.

3.8.4.1.1.3.13 Global Stability Analysis

During its review of the Shield Building Report, the staff identified that the applicant had not addressed global stability of the shield building. The global stability issue was discussed and identified under Action Item 6 at the June 9-11, 2010, meeting.

To address Action Item 6, the applicant provided an analysis of global stability in its letter dated June 30, 2010. The applicant concentrated on demonstrating that the PCCWST does not add significant weight to the structure and that the long-term effects of creep are negligible. As such, the cylindrical wall was analyzed for stability under hoop and axial compression. The applicant reported that the compressive stress resulting from the dead weight of the structure was [ ]. Consequently, the applicant stated that because the dead weight stress is small the effects of creep are negligible. The applicant performed an analysis for axial buckling and calculated that the elastic buckling compressive stress was [ ]. Because the concrete compressive stress is [ ], the applicant concluded that the concrete would crush before buckling occurred.

The staff reviewed the applicant’s technical basis for global stability and found it to be consistent with the ACI Committee 334 report, “Concrete Shell Structures Practice and Commentary.” The staff found the analysis to be acceptable based on an independent calculation of the critical buckling strength of elastic shells under compressive loads.

Pushover Analysis

The applicant performed nonlinear confirmatory analysis to predict the behavior of the shield building up to and beyond design basis seismic loading and assess the potential for collapse. The applicant used its [ ] model of the nuclear island to perform a nonlinear pushover analysis of the shield building. The model included the shield building and the entire auxiliary building. This finite element model did not impose constraints that would force a mode of deformation of the shield building structure. Using this model, the applicant's analysis tracked tensile stresses and strains in the steel faceplates, in-plane and out-of-plane shear deformations and stresses, stresses and strains in the [ ], deformations in the connection regions and stresses and strains in the [ ] in the RC wall below the SC wall. The applicant's analysis explicitly modeled the interaction of the shield building with the roof and walls of the auxiliary building. The applicant's model also did not exclude the possibility of shear failures. Instead, it considered concrete cracking for out-of-plane loads as well as in-plane loads and the subsequent distribution of forces to the steel reinforcement. Since the applicant's verification and validation of the model against its own test data did not capture brittle failures, the applicant tracked the possibility of local onset of such brittle shear failures through the use of limiting strains in the [ ] as well as through the combined use of analysis methods with increasing refinement, that is, the combination of [ ] models.

For its analysis, [ ]. In addition, the applicant considered various combinations of the directions and intensity of the seismic loads in the two horizontal directions and in the vertical direction. Under these loading conditions and without constraints in the response modes of the structure the applicant calculated the response of the structure to proportionally increasing loads. Proportional increase of the loads is an approximation in a static pushover analysis. As the structure yields and the response becomes increasingly inelastic, there is a potential for redistribution of the loads through the height of the structure that may affect the subsequent response mode of the structure. The results of the applicant's analysis show that significant inelastic behavior of the wall, other than concrete cracking, will not occur at the design basis loads and will only start at loads closer to the review level earthquake (RLE). On this basis, loading conditions that deviate significantly from those used by the applicant are not expected up to the SSE and RLE levels.

The applicant's analysis results showed that the highly stressed regions of the shield building were near structural discontinuities such as the connection to the basemat at the 30.40 m (100 ft) elevation, in the region above the roof of the auxiliary building and at the connection of the SC wall to the RC walls. The analysis predicts yielding initiation through yielding of the [ ].

The results of the pushover analysis confirm that the shield building stresses, strains and deformations remain small at the design basis loads and that significant yielding in the SC wall does not start until loading levels beyond the SSE and of the order of the RLE. The results of the analysis confirm that the high stress areas of the wall with complex states of stress from the combination of high membrane forces and out-of-plane forces are the areas of the wall for which [ ], described in Section 4.3.5.2 of this report, showed that these models exhibit ductile out-of-plane behavior under cyclic loading.

As a result of the above global stability calculation and confirmatory pushover analysis, the staff considers the issue of global stability and related Action Item 6 to be resolved.

3.8.4.1.1.3.14 Construction and Inspection Methods

The staff had concerns about the construction and inspection methods that the applicant had planned to use to ensure the integrity and safety of the shield building design. The staff's concerns centered on the sequence of construction and considerations for the wet concrete loads, thermal loads, and welding processes to be used. The staff was also concerned about how the applicant would inspect for voids, cracking, delaminating, and substandard construction of concrete. During a meeting on February 23, 2010, the staff raised concerns related to the use of a qualified inspector in accordance with the ACI 318 Code and the need for continuous inspection throughout construction.

As indicated in Section 9.2 of the Shield Building Report, the applicant plans to construct the shield building in an alternating sequence with the construction of the CV. After setting the first ring of the CV, approximately 12.2 m (40 ft) high, the shield building modules will be installed and filled with 3 m (10 ft) concrete lifts. To help ensure the integrity of the design of the shield building, the applicant will undertake a mockup program focused on three critical areas:

(1) the vertical RC-to-SC connection
(2) the horizontal RC-to-SC connection
(3) the air-inlet/tension ring structure

The results of the mockup program will be used to gain insights into any modifications to the design that may be needed before construction.

In Section 9.5 of the Shield Building Report, the applicant specified that the welding codes and process and welding inspection criteria for structural welding are in accordance with ANSI/AISC N690 and AWS D1.1, “Structural Welding Code—Steel.” In Section 9.6 of the Shield Building Report, the applicant specified that ANSI/ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications,” as well as ANSI/AISC N690 and AWS D1.1, govern the design requirements for the fabrication, assembly, and installation of the SC wall module components and construction inspection.

The staff is concerned that the proposed SC/RC connection and the tension ring/air-inlet connection may have constructability problems, such as steel rod alignment, aggregate size, air entrapment, and bleed water accumulation. Further, the staff is concerned that the proposed connection may have design implications, such as elongation in the reinforcing bars, shear friction transfer, and compression force transfer. The goal is to increase the confidence that the success of carefully designed mock-up tests would be replicated during construction.

In particular, the staff believes that concrete placement plans for the SC and RC connection region, tension ring and air inlet should be fully developed with emphasis on ensuring venting of air and complete filling of cavities. The applicant states in Revision 2 of the Shield Building Report that horizontal construction joints at the top of each concrete placement, including those near the bottom of the ring girder, would be prepared in accordance with ACI 349, Article 11.7.9. Since this reference does not specify a preparation procedure, the applicant should prepare one as the construction plans progress.

With respect to staff concerns raised about the method of inspecting the SC wall module given that the design includes concrete between two steel plates without visual access, in Section 9.8 of the Shield Building Report, the applicant evaluated several nondestructive examination (NDE) technologies for their potential for determining concrete defects and proposed to use the [

]. The [ ] approach is acceptable to the staff when used in conjunction with acceptance criteria for defects that would trigger more detailed evaluations when necessary. In Section 9.8 of Revision 2, the applicant developed criteria for acceptable levels of defects, and in Table 9.8-2, criteria for spacing between defects.

The staff understands that the spacing of defects [ ], both the maximum spacing and the spacing used for acceptance; involve both horizontal and vertical dimensions and not just a single linear dimension. On page 33 of Revision 2, the applicant wrote that a 95/95 sampling methodology would require a random grid of 59 total sampling point locations in each of the three critical areas of the inner shield building. The staff understands this to mean that for each sampling scan in each of these critical areas, 59 sampling points would be required, and not to mean that the inspection would consist of only three scans, one per critical area, and each with 59 sampling points. The staff notes that the applicant did not provide in Revision 2 specific technologies for the more detailed evaluations when acceptance criteria are not met. Finalized inspection procedures should include those technologies. On page 9-33 of the Revision 2 report, the applicant wrote that if inspection ports cut in the steel plates become necessary for NDE, the location of those ports would be at those sample point locations. The staff notes that the Revision 2 report does not indicate if a location of inspection points is a single point location or a grid of test points. This needs to be specified in the completed inspection program.

Based on its review, the staff found that the applicant has addressed the staff's concerns. Particularly, the applicant has described the construction sequence; and the use of mock-ups in order to help ensure the integrity of the designed structure during construction. However, the staff believes that the applicant should complete its development of all construction and inspection implementation procedures, establish the QA/quality control procedures, finalize its selection of the NDE technology, and determine a method to help ensure that the results of the mock-up program and the qualification of the inspectors are implemented at the site. This topic is discussed further in Section 3.8.4.1.1.4 below.

3.8.4.1.1.4 Inspections, Tests, Analyses, and Acceptance Criteria

AP1000 DCD, Revision 17, Tier 1, Table 3.3-6 addresses the NI structures, including the critical sections. The acceptance criteria require a report that reconciles deviations during construction and concludes that the as-built shield building structures, including critical sections, conform to the design-basis loads without loss of structural integrity or the safety function. The staff finds that the AP1000 DCD Tier 1 ITAAC included sufficient requirements for the design acceptance of the shield building and its critical sections. Hence, the staff did not identify any additional ITAAC based on its review of the shield building design.

In Chapter 9 of the Shield Building Report, the applicant described the construction and inspection methods for the shield building. The staff's review found that the applicant must provide a COL information item to ensure that the shield building is constructed as designed to perform its intended safety function.

In RAI-SRP3.8.4-SEB1-04, the staff asked the applicant to provide commitments for unique construction and inspection procedures, such that the COL applicant will develop and follow procedures described in the COL information item. Further, the staff requested that the COL information item include the construction sequence, mockup requirements for the critical areas of the shield building, concrete placement methods, inspection of modules before and after concrete placement, and QA procedures.

In its response dated September 3, 2010, the applicant proposed a new COL information item including construction procedures and inspection procedures for SC construction. The applicant stated that these procedures derive from Chapter 9 of the Shield Building Report and will be added to AP1000 DCD Section 3.8. Further, the applicant stated that for SC construction, the construction inspection will be done in accordance with the applicable codes and standards listed in AP1000 DCD Section 3.8.4.2. For the shield building mockup program, the applicant proposed to use the heavily reinforced sections, which are deemed to be the sections of the design that present difficult construction issues. These sections include the lower section of RC/SC interface, horizontal RC/SC connection, and the air-inlet structure/tension ring. Additionally, the applicant stated that similar mockups will also be performed for the SC module and that insights from these mockups will be applied in construction.

The COL information item states that COL holders referencing the AP1000 DC will develop construction and inspection procedures to implement the commitments for concrete-filled steel plate modules. Further, these procedures will address concrete placement, use of construction mockups, and inspection of modules before and after concrete placement.

The staff reviewed the response to RAI-SRP3.8.4-SEB1-04 and the proposed COL information item and finds that the applicant's commitment to perform shield building mockups and develop construction and inspection procedures is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and table, which resolve this issue.

3.8.4.1.1.5 Shield Building Conclusion

The staff evaluated the adequacy of the design of the shield building, as provided by the applicant in the Shield Building Report dated May 7, 2010, and as supplemented by 2010 submittals dated June 24, June 30, July 30, and September 3, and a June 15, 3011 submittal, and finds that the design of the shield building meets the relevant requirements of the regulations as provided in 10 CFR 50.55a and GDC 1 and 2 of Appendix A to 10 CFR Part 50.

Based on its evaluation, the staff finds that the design of the shield building demonstrates reasonable assurance that it will perform its intended safety function, and, therefore, is acceptable. Moreover, the staff finds that the shield building is adequately designed to withstand the effects of natural phenomena, thereby ensuring it will perform its intended safety function.

The staff recognizes that design standards or industry codes specific to the design of the SC wall module do not exist in the United States. However, the staff finds that the applicant used an alternative approach and implemented an integrated design methodology, including design, analysis, confirmatory analysis, testing, construction, and inspection, applicable for the development of the design of the AP1000 shield building. Specifically, the design methodology uses ACI 349 for RC design and supplements it with confirmatory analysis and confirmatory testing for its application to the AP1000 design of the SC wall module. Specifically, for the design of the SC cylindrical wall, air inlets, and SC/RC connection, the ACI 349 methodology was used for the design and the applicant supplemented its design with confirmatory analysis and testing. In view of the integrated methodology adopted for the shield building design, the staff believes the applicant's alternative approach is acceptable for this first-of-a-kind engineering design.

In addition, the staff finds that the applicant's modifications to improve the original design of the shield building, such as the use of the [ ] in the SC wall module and enhancements to the SC-to-RC and basemat connections, the roof, and tension ring/ring girder and air-inlet regions, make significant improvements in the design. Specifically, the applicant's inclusion of [ ] significantly improves the capacity of the SC wall module and enables the structure to function as a unit under design-basis loads. Further, the staff finds that the design possesses the basic elements of strength, stiffness, and ductility. The revised SC-to-RC connection allows for a [ ], while the revisions to the design of the tension ring and air-inlet region significantly improve the load path and thus, the transfer of forces.

The applicant's analysis of strength and ductility is acceptable for SSE demand, and the use of confirmatory tests in conjunction with confirmatory analysis demonstrates that the capacity based on ACI 349 equations for the design of SC structures is adequate to meet the SSE demands. With regard to the analysis supporting the design of the shield building, the applicant performed three levels of analysis to determine the load magnitudes, response spectra and member forces and the required design strength in accordance with the ACI 349 Code. In addition, the applicant's consideration of thermal effects, fatigue, creep, and construction loads in the design of the shield building were reasonably well supported by modeling and detailed confirmatory analyses.

As part of the integrated design methodology, the applicant conducted confirmatory tests of the SC wall module to confirm the adequacy of those portions of the AP1000 shield building design that fall outside the scope of existing design codes and to demonstrate the level of conservatism in using ACI 349. Specifically, [ ] resulted in demonstrating the desired ductile behavior, and the out-of-plane shear test with [ ]. In addition, the [ ] of the SC wall module indicated substantial strength margin to design loads, but the module was not tested to capacity; therefore, the test did not demonstrate that the SC module would not fail in a brittle manner under cyclic loading. In a report referenced by the applicant, the staff found that a Japanese test of scaled models of SC structures (with geometry similar to the AP1000 shield building design) had demonstrated sufficient ductility for cyclic in-plane shear loading. However, the Japanese tests were not performed for cyclic out-of-plane shear loading.

The applicant addressed ductility for out-of-plane loading by referencing ACI 349, Article 21, pertaining to moment-resisting frames. The staff finds that ACI 349 (Article 21) is intended for moment frame structures and is not directly applicable to cylindrical shell structures, such as the AP1000 shield building. Cylindrical shells will distribute forces in a manner that differs from a two- or three-dimensional framed structure. However, the staff agrees that the inherent premise used in ACI 349, Article 21, of providing ductile detailing where demands are high, can be extended to a cylindrical shell if analysis has been performed to identify locations of high demands.

The staff finds that to resist out-of-plane shear loading, the shield building design uses [ ] to ensure that the SC modules will function as a unit. For the regions of the SC wall module with higher out-of-plane shear loads, and where yielding of the SC wall module would be expected to initiate under a combination of tensile forces and out-of-plane bending for seismic loads, the applicant detailed the SC modules with [ ] to provide out-of-plane shear ductility. For the regions of the SC wall with low out-of-plane shear demands and [ ], the SC wall detailing does not provide out-of-plane shear ductility based on the test results. In these regions, the out-of-plane shear demands calculated by the applicant are low, and the SC wall modules as detailed provide conservative strength demand-to-capacity ratios. Based on: (1) demonstration of conservative strength and adequate cyclic behavior for the SC module with [ ] ; (2) confirmatory analysis that identified locations of potential SC steel plate yielding; and (3) the analogy with ACI 349, Articles 21.3 and 21.4, which require ductile detailing only where demands are high and plastic hinges are expected to form, the staff finds the applicant's use of [ ] at [ ] spacing to be acceptable.

Furthermore, the staff finds SC module design is acceptable on the basis that the applicant demonstrated that its lowest margin is 18 percent (in-plane shear) under design-basis SSE loads and on the staff's determination that other SC modules with design characteristics similar to the AP1000 shield building possessed sufficient ductility under in-plane shear cyclic loading. Regarding out-of-plane shear loading of the SC module with [ ], the staff finds that although these specimens failed in a brittle manner, there is significant margin between the failure loads of the two test specimens [ ] and the maximum SSE demand of [ ]. Lastly, the applicant's construction and inspection processes involving the use of mock-ups for two key areas, the SC-to-RC connection and the ring girder-to-SC connection, are acceptable, although the staff finds that the applicant should finalize its implementation of its construction and inspection procedures and methods. The applicant should also determine a method to help ensure that the results of the mock-up program are correctly implemented at the site.

In summary, based on the above discussions, the staff finds that the design of the AP1000 shield building is acceptable.

3.8.4.2 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 as they relate to other seismic Category I structures as documented in AP1000 DCD Revision 19, against the relevant acceptance criteria as listed above and in NUREG-0800 Section 3.8.4.

In subsequent revisions to APP-1200-S3R-003, "Design Report for the AP1000 Enhanced Shield Building," the applicant made appropriate changes to the report. Based on the review of these changes, staff concludes that APP-1200-S3R-003, Revision 4, is acceptable because the analyses and design were performed in accordance with the ACI 349 Code, applicable RGs, and NUREG-0800, Section 3.8.4.

The staff concludes that the design of the other seismic Category I structures meets all applicable acceptance criteria. In summary, based on the above discussions, the staff finds that the design of other seismic Category I structures including the AP1000 shield building is acceptable.

The applicant proposed to amend the existing design certification rule, in part, to address the requirements of the aircraft impact assessment (AIA) rule. The AIA rule itself mandated that a design certification rule (DCR) be revised (either during the DCR's current term or no later than its renewal) to address the requirements of the AIA rule. In addition, the AIA rule provided that any combined license issued after the effective date of the final AIA rule must reference a DCR complying with the AIA rule, or itself demonstrate compliance with the AIA rule. The AIA rule may therefore be regarded as inconsistent with the finality provisions in 10 CFR 52.63(a) and Section VI of the AP1000 DCR. However, the NRC provided an administrative exemption from these finality requirements when the final AIA rule was issued. See June 12, 2009; 74 FR 28112, at 28143-45. Therefore, the NRC has already addressed the finality provisions of applying the AIA rule to the AP1000 with respect to the AP1000 and referencing COL applicants.

3.8.5 Foundations

Using the regulatory guidance in NUREG-0800 Section 3.8.5, "Foundations," the staff reviewed areas related to the foundations of all seismic Category I structures. The specific areas of review provided in NUREG-0800 Section 3.8.5 are as follows: (1) description of the foundations; (2) applicable codes, standards, and specifications; (3) loads and load combinations; (4) design and analysis procedures; (5) structural acceptance criteria; (6) materials, quality control, and special construction techniques; (7) testing and inservice surveillance programs; (8) ITAAC; and (9) COL action items and certification requirements and restrictions. Not all of these areas were applicable to the review of the proposed changes to AP1000 DCD Section 3.8.5 and the following SER sections provide the staff's evaluation for the relevant areas.

In its previous evaluations of AP1000 DCD, Section 3.8.5, the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR 50.55a, "Codes and Standards"; 10 CFR Part 50, Appendix A, GDC 1; GDC 2; and GDC 4. The staff found that the design of the AP1000 foundations was in compliance with these requirements, as referenced in NUREG-0800 Section 3.8.5 and determined that the design of the AP1000 foundations, as documented in the AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

In the AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.5 of the certified design:

1. As a result of the extension of the AP1000 design from just hard rock sites to sites ranging from soft soils to hard rock, various seismic re-analyses of the NI structures were performed. Whereas the original design relied upon the equivalent static method of analysis for seismic loading, the re-analyses included the additional use of response spectrum and time history methods of analysis. Appendix G of DCD Revision 17 indicates that the response spectrum analysis was used for the 3D refined finite element model of the NI and for the analysis of the PCS valve room and miscellaneous steel-framed structures, flexible walls, and floors. Time history analyses were used to determine maximum soil bearing

pressures under the NI and, subsequent to the submittal of DCD Revision 17, to perform an updated NI stability evaluation.

2. In DCD Revision 16, the applicant revised Section 3.8.5.4.1 - Analyses for Loads during Operation, regarding the reinforcing steel under the shield building and the auxiliary building. Additional reinforcement is provided in the design of the basemat for soil sites such that the basemat can resist loads 20 percent greater than the demand calculated using the equivalent static acceleration analyses on uniform soil springs. The design accommodates potential site specific soil variability beneath the basemat in the horizontal (lateral) directions.

3. In DCD Revision 16, the applicant included in Section 3.8.5.4.2 a description of the analyses which evaluate the effects of different construction sequences on settlement and the design of the basemat. DCD Revision 17 made some additional revisions to describe the concrete placement sequence in the basemat and in the auxiliary building during construction.

4. In DCD Revision 16, the applicant revised Section 3.8.5.4.3 - Design Summary Report. DCD Revision 15 indicated that the results of the evaluation will be documented in an as-built summary report by the COL applicant. In DCD Revision 16, this was revised to state, “The results of the evaluation will be documented in an as-built summary report.”

5. In DCD Revision 16, the applicant revised Section 3.8.5.4.4 - Design Summary of Critical Sections. The design approach of the basemat for two of the critical sections was revised to design these sections as two way slabs.

6. In DCD Revisions 16 and 17, several revisions were made in Section 3.8.5.5 - Structural Criteria, regarding the sliding and overturning stability evaluations. In DCD Revision 16, Section 3.8.5.5.3 - Sliding, the sliding coefficient of friction between the basemat and the soil was revised from 0.55 to 0.70. In DCD Revision 17, Section 3.8.5.5.4 - Overturning, the equation used to calculate the factor of safety for overturning due to the safe shutdown earthquake was revised.

7. In DCD Revision 16, the applicant revised Section 3.8.5.6 - Materials, Quality Control, and Special Construction Techniques. DCD Revision 15 indicated that the COL applicant would provide information related to the excavation, backfill, and mudmat. In DCD Revision 16, this was revised to state that Section 2.5.4.5.3 describes the information related to the excavation, backfill, and mudmat.

8. In DCD Revision 16, the applicant revised Section 3.8.5.7 - In-Service Testing and Inspection Requirements. DCD Revision 15 indicated that the COL applicant has the responsibility to determine the need for foundation settlement monitoring. In DCD Revision 16, this was revised to state that the need for foundation settlement monitoring is site-specific as discussed in subsection 2.5.4.5.10.

The evaluation of changes to the description of foundations, applicable codes, standards, and specifications, loads and load combinations, and the design and analysis procedures may be found in the evaluation of TR-85, presented below.

3.8.5.1 Nuclear Island Basemat Technical Report TR-85

Since the AP1000 design was previously certified for use at an HR site, the applicant submitted TR-85, Revision 0, to summarize the design of the NI basemat and exterior walls below grade for both HR and soil sites. This report also describes interface demands to be satisfied at a site. TR-85 Revision 0 indicates that the report also provides an updated baseline for the as-designed configuration and validates the basemat and foundation design against the updated seismic spectra and soil foundation conditions. TR-85 was subsequently modified in Revision 1 to address a number of the outstanding RAIs. Some of the information in TR-85 is included in the AP1000 DCD, Revision 17.

As a result of the staff's review of TR-85, a number of RAIs were sent to the applicant. Based on these RAIs, the applicant made a number of revisions in the analyses and design methods to address the issues raised. The description provided below presents the staff's evaluation of the key issues.

3.8.5.1.1 Design of NI Walls below Grade

As a result of the staff's review of TR-85, a number of questions were identified related to the design of the foundation walls below grade. These questions were captured in RAI-TR85-SEB1-02, RAI-TR85-SEB1-04, RAI-TR85-SEB1-34, and RAI-TR85-SEB1-40. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address the issues raised. The description provided below presents the staff's evaluation of the key issues related to the design of the foundation walls below grade.

As described in the applicant's response to RAI-TR85-SEB1-02, the analytical approach to calculate the pressure loads on the side walls below grade (embedded walls) consisted of hydrostatic pressure from ground water, at rest earth pressure, surcharge pressure, dynamic earth pressure, and passive earth pressure. The seismic earth pressure was calculated in accordance with ASCE 4-98, Section 3.5.3, which utilizes the elastic solution for dynamic soil pressures. In addition to designing the foundation walls to the seismic earth pressure, the RAI response also indicates that the NI exterior walls are designed for the passive soil pressure in the load combinations that include SSE.

The staff finds that the approaches used by the applicant to calculate these various soil pressure loads were in accordance with industry-wide soil mechanics methods and were consistent with the criteria presented in NUREG-0800 Section 3.7 for seismic loads and Section 3.8 for design methods, and, therefore, are acceptable.

3.8.5.1.2 Maximum Soil Bearing Pressure beneath the Basemat during SSE

As a result of the staff's review of TR-85, a number of questions were identified related to the calculation of the maximum soil bearing pressures beneath the basemat due to the SSE. These questions related to soil bearing pressure were captured in RAI-TR85-SEB1-03, RAI-TR85-SEB1-04, RAI-TR85-SEB1-06, RAI-TR85-SEB1-15, RAI-TR85-SEB1-26, and RAI-TR85-SEB1-40. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address these issues. The description provided below presents the staff's evaluation of the key issues related to the soil bearing pressure evaluations.

Based on the response provided to RAI-TR85-SEB1-03, the maximum dynamic bearing pressure on soils resulting from SSE was 5745.6 kPa (120,000 pounds per square foot (psf)) for the HR case in the previous AP1000 certified design using the more conservative equivalent static analysis method. The 5.746 MPa (120,000 psf) pressure was reduced to 1.331 MPa (27,800 psf) for the HR case by using a more realistic 2D [ ] nonlinear (liftoff) analysis. The 2D [ ] nonlinear (liftoff) analysis showed that the SM soil case gives a somewhat higher dynamic bearing pressure, 1.652 MPa (34,500 psf), than that of the HR case. The applicant also calculated the maximum dynamic bearing pressure on soils by using the [ ] 3D finite element NI20 model with a seismic time history SSI analysis. This analysis was performed for the HR case and five soil conditions, and the resulting maximum dynamic bearing pressure is 1.676 MPa (35,000 psf). This analysis is described in detail in Section 2.4.3 of TR-85, Revision 1, and TR-03 (November 2008). The maximum soil bearing pressure demand of 1.676 MPa (35,000 psf) for the NI is presented in AP1000 DCD Tier 1, Section 5.0, "Site Parameters." The applicant also explained how the time history analyses removed a number of conservatisms inherent in the equivalent static seismic analysis, which led to the large reduction in the soil bearing pressure. Based on this explanation and the use of a more accurate [ ] 3D finite element NI20 model analysis, which was also confirmed with the independent 2D nonlinear liftoff [ ] analysis, the staff concludes that the applicant has used proper methods to obtain the maximum dynamic bearing pressure on the soil.

3.8.5.1.3 Stability Analysis (Sliding and Overturning) of the Basemat and Foundation Waterproofing Systems

As a result of the staff's review of TR-85, a number of questions were identified related to the calculation of the stability analysis of the NI basemat and the foundation waterproofing systems. These questions were captured in RAI-TR85-SEB1-04, RAI-TR85-SEB1-07, RAI-TR85-SEB1-10, RAI-TR85-SEB1-11, RAI-TR85-SEB1-34, RAI-TR85-SEB1-35, and RAI-TR85-SEB1-40. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address these issues. The description provided below presents the staff's evaluation of the key issues related to the stability evaluations.

Based on the response to RAI-TR85-SEB1-10, for the overturning and sliding stability evaluation, the applicant initially used the 3D [ ] NI20 model. For the SSE loading, an equivalent static analysis was performed and demonstrated that without the use of passive soil pressure resistance, the overturning factors of safety were met. However, for sliding, difficulties were identified in satisfying the sliding factor of safety. Therefore, the applicant performed another more realistic nonlinear analysis with sliding friction elements using a modified 2D [ ] model that was used previously to study the basemat uplift. This model, which is described in Section 2.4.2 of TR-85, was modified to use sliding friction elements at the interface of the basemat and the soil. The model considered basemat vertical uplift in addition to sliding. A direct integration time history analysis using the modified 2D [ ] model was performed to evaluate the basemat stability issue. Three soil cases that have the lowest factor of safety-related to sliding were evaluated. These three cases are HR soil, UBSM soil, and SM soil. The seismic input was increased by 10 percent so as to maintain the factor of safety against sliding of 1.1. No passive soil resistance was considered in the analyses. The resulting maximum deflection at the base using a coefficient of friction of 0.55 was 0.08 cm (0.03 in) for all three soil cases. This horizontal sliding deflection was considered to be negligible and no passive soil pressure resistance was necessary from the backfill. Therefore, the applicant concluded that the NI is stable against sliding and there is no passive pressure required to maintain stability. The AP1000 DCD requires COL applicants to demonstrate by testing that soils beneath their basemat possess a minimum coefficient of friction of 0.7, which is equivalent to the soil friction angle of 35 degrees, and this provides additional conservatism for the basemat against sliding stability.

The staff's review of the RAI-TR85-SEB1-10 response related to the seismic stability evaluation of the NI concludes that the overall 2D [ ] nonlinear sliding analysis approach appears to be appropriate; nevertheless, a review of the applicant's calculation was needed to confirm the proper implementation of this methodology is appropriate. At the seismic audit conducted during the week of June 14, 2010, the staff reviewed the 2D [ ] non-linear sliding stability evaluation. As a result of this review a change was made to the [ ] sliding/contact finite element that resulted in larger horizontal displacements. The resulting maximum displacement at the base of the NI basemat was 0.356 cm (0.14 in) without buoyant force consideration, and 0.61 cm (0.24 in) with buoyant force effects. These values are larger than the previously reported results, 0.76 mm (0.03 in) without buoyant force and 1.14 mm (0.045 in) with buoyant force effects. However, these values are still judged to be negligibly small, especially when the conservative analysis approach of neglecting sliding resistance from the soil passage pressure and neglecting the additional fictional forces along the barrier portions of the NI side walls are considered. Therefore, it is concluded that the NI is stable against sliding. However, the staff notes the need to revise the response to RAI-TR85-SEB1-10 to reflect the revised finite element for sliding and the increase in displacements, and provide the DCD and TR-85 changes to reflect the sliding evaluation.

Since wind and tornados generate less horizontal sliding force and overturning bending moment than the SSE does, the applicant concluded that the NI, which does not have stability problems against SSE, will not have problems against wind and tornados.

As a result of the staff's structural audit conducted during the week of August 10, 2009, the staff requested justification as to why TR-85 is not identified as Tier 2* since it is referenced in AP1000 DCD Section 3.8.5 and it includes key details of the design of the foundation. Similarly, justification was not provided for identification of Tier 2* for TR-09, TR-57, and the updated shield building reports. Therefore, in a follow-up to RAI-TR85-SEB1-10, the staff requested that TR-09, TR-57, and TR-85 be identified as Tier 2* information in the AP1000 DCD, or an acceptable justification be provided.

At the seismic audit conducted during the week of June 14, 2010, the staff reviewed the 2D [ ] nonlinear sliding stability evaluation. As a result of this review, a change was made to the [ ] sliding/contact finite element, which resulted in larger horizontal displacements. The resulting maximum displacements, reported in the applicant's letter dated August 25, 2010, at the base of the NI basemat were determined to be 0.30 cm (0.12 in) without buoyant force consideration, and 0.48 cm (0.19 in) with buoyant force effects considered. These values are larger than the previously reported results of 0.77 mm (0.03 in) without buoyant force consideration, and 1.14 mm (0.045 in) with buoyant force effects. However, these values are still judged to be negligibly small, especially when the conservative analysis approach of neglecting any sliding resistance from the soil passive pressure and neglecting the additional frictional forces along the buried portions of the NI side walls are considered. Therefore, it can be concluded that the NI is stable against sliding. However, the applicant must revise the response to RAI-TR85-SEB1-10 to reflect the revised finite element for sliding and the increase in displacements, and provide the mark-ups for the AP1000 DCD changes and TR-85 to reflect the changes in the sliding evaluation.

In response to the above requests, the applicant's letters dated July 30, 2010, and August 25, 2010, indicated that the applicant would review the information in the RAI responses and the structural TRs for the key analysis and design information that should be included in the AP1000 DCD, and would provide DCD mark-ups for the complete Sections 3.7 and 3.8, as well as Appendixes 3G, 3H and 3I, identifying the Tier 2* information. In addition, the applicant provided the mark-ups for the AP1000 DCD and TR-85 to reflect the changes in the sliding evaluation due to modifications for the sliding/contact finite element. The staff's review of the RAI responses in the two letters concluded that the proposed approach, to add the specific Tier 2* information from the applicable TRs and shield building report(s) to the AP1000 DCD, is acceptable because mark-ups will be provided and give the staff an opportunity to confirm that the required information will be identified as Tier 2* in the AP1000 DCD. The response regarding the revised NI seismic sliding evaluation is also acceptable because it provides the mark-ups for the changes to the AP1000 DCD and TR-85 to reflect the changes in the sliding evaluation and the increases in seismic displacement due to sliding. The staff notes that the applicant clarified the design basis by letters dated October 21, 2010, whereby they withdrew TR-57 and provided mark-ups of the DCD to show the removal of references to TR-57 and stated the location where the information, as updated, appears in the proposed DCD and an appendix thereto. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, tables and figures and TR-85 report. In addition, to document proprietary design detail information, including Tier 2* information, for the AP1000 Shield Building cylinder and connections to the auxiliary building and basemat, the applicant proposed a proprietary report, APP-GW-GLR-602, entitled "AP1000 Shield Building Design Details for Select Wall and RC/SC Connections." The staff reviewed APP-GW-GLR-602 and determined that it resolved the issues discussed above and is, therefore, acceptable.

A concrete mud mat consisting of an upper and a lower mud mat is placed on top of the soil foundations to provide a level support for the structural concrete basemat. A waterproofing membrane is placed between the upper mud mat and the lower mud mat. In RAI-TR85-SEB1-35, the staff requested that the applicant describe, in greater detail, the types of waterproofing materials to be used and how the coefficient of friction for these materials, assumed in the sliding stability evaluations, will be demonstrated. In response, the applicant explained that one of three types of waterproofing systems is used: plasticized polyvinyl chloride (PVC) membrane, HDPE membrane, or a crystalline spray type material. The AP1000 DCD requires COL applicants to demonstrate by testing that the waterproofing membrane will achieve a minimum coefficient of friction of 0.55 (the value which was used for the NI sliding stability analysis) between it and the concrete mud mat.

The staff's review of the applicant's responses to RAI-TR85-SEB1-35 determined that the information provided to describe the waterproofing materials was not sufficient and that further revisions in the AP1000 DCD were required to reflect the revised details of the waterproofing materials. The remaining items that needed to be addressed relate to the proposed mark-up in the AP1000 DCD describing the waterproofing materials, more detailed information about the type and industry standards used for the waterproofing membrane, and information that demonstrates the adequacy of the crystalline waterproofing material.

In the applicant's letter dated June 30, 2010, the response to RAI-TR85-SEB1-35 indicated that the waterproofing system for the below grade walls and mud mat would consist of either the HDPE double-sided textured membrane; HDPE single-sided adhering sheet membrane; self-adhesive, rubberized asphalt/polyethylene membrane (for walls only); or sprayed-on waterproofing membrane based on polymer-modified asphalt or polyurea. The response explained that the use of the crystalline waterproofing material had been eliminated as an option. In addition, the industry standards used to specify performance requirements and other design requirements (e.g., maximum crack width) for the waterproofing systems were provided.

The proposed mark-ups to the AP1000 DCD describing the waterproofing materials and performance requirements were also provided and found to be acceptable based on the use of the applicable industry standards and industry practices. Also, the elimination of the use of the crystalline material resolves the questions raised regarding the adequacy of this material. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

3.8.5.1.4 The Effect of Basemat Liftoff from the Ground

Section 2.4.2 of TR-85, Revision 1, and the response to RAI-TR85-SEB1-14 described analyses performed using a 2D [ ] nonlinear model to evaluate the potential effects of liftoff. This was needed because [ ] analyses cannot model nonlinear behavior, such as liftoff of the NI structure from the soil. The [ ] analyses permit tension to be transferred across the interface between the basemat and the soil. Therefore, analyses were performed with the 2D [ ] nonlinear model, which allowed for liftoff, and the results were compared to 2D [ ] analyses, which do not have liftoff. The NI superstructures (i.e., structures above the basemat) were represented as stick models in both the 2D [ ] model and the 2D [ ] model. In the 2D [ ] model, the soil was represented by horizontal and vertical springs. The springs were only effective when the basemat was in contact with the soil (i.e., when the springs were in compression).

The results of the two analyses were compared in terms of FRS in the structures, member forces, and soil bearing pressures. The applicant provided comparisons of in-structure FRS, member forces and soil bearing pressures. The applicant indicated that these comparisons show that there is no significant difference between the 2D [ ] nonlinear analyses and the 2D [ ] linear analyses. On this basis the applicant concluded that the NI superstructure may be designed neglecting liftoff, but the basemat design does need to consider the effects of liftoff. Thus, Section 2.6 in TR-85 provides the analysis and design of the NI basemat, which uses a 3D [ ] model that does consider liftoff.

The staff review of the tabulated comparisons of the member forces at representative locations between the 2D [ ] and the 2D [ ] analyses showed a maximum difference of 2.7 percent. The in-structure generated response spectra comparisons at key locations showed that the 2D [ ] nonlinear analysis spectra were often below or within about 10 percent above the 2D [ ] linear results, except at the very low frequency of about 4.8 Hz in the vertical direction where the difference is about 15 percent. For soil bearing comparisons, the differences for the maximum soil bearing pressures were within about 6 percent. Since the applicant performed a nonlinear [ ] analysis with liftoff capability and showed that the results are reasonably close to the [ ] results without liftoff capability, the staff finds the applicant’s approach for addressing the NI liftoff effects acceptable. Therefore, RAI-TR85-SEB1-14 is resolved.

3.8.5.1.5 Basemat Design

3.8.5.1.5.1 Seismic Analysis of NI Basemat and Soil Reaction Force (Pressure) at the Bottom of the Basemat

The seismic analysis was based on the 3D [ ] finite element NI05 model using seismic equivalent static accelerations, which were obtained from the time history analysis of the NI on HR, prior to the design changes made to enhance the shield building. This 3D [ ] NI05 analysis of the basemat is described in Section 2.6.1 of TR-85, Revision 1, and in the responses to RAI-TR85-SEB1-21, RAI-TR85-SEB1-22, and RAI-TR85-SEB1-23. The model is nonlinear because soil springs can only take compression but not tension when the basemat lifts off the ground. To verify the adequacy of the equivalent static accelerations used in the 3D [ ] NI05 model another study was performed. First, a linear analysis using the equivalent static accelerations discussed above was performed to determine the total base reactions and soil bearing pressures. Then, a time history fixed base analysis, which accounted for the various soil profiles, was performed. The time history inputs for this analysis were developed based on the envelope of the basemat responses given by the 3D [ ] analyses. The 3D [ ] analyses considered five soil cases: FR, SR, UBSM, SM, and SS. Based on the comparison of the base reactions and soil bearing pressures from the equivalent static analysis (for the HR condition) and the time history analysis (for the range of soil conditions), the applicant concluded that the study demonstrated that the equivalent static accelerations from the prior time history analysis of the NI on HR, are still acceptable.

The staff finds that the 3D [ ] NI05 model is appropriate since it was developed in accordance with industry methods and is consistent with the guidance presented in NUREG-0800 Section 3.8.5. The applicant's use of the equivalent static analysis as described above is reasonable because the applicant compared the base reactions from the NI and soil bearing pressures obtained from the equivalent static analysis with the results from the time history analysis that considered the range of possible soil conditions.

The soil pressure imposed on the bottom of the basemat, obtained from the above seismic analyses, is based on the assumption that the NI rests on a uniform soil site. For a site to be considered uniform, the variation of Vs in the material below the foundation to a depth of 36.7 m (120 ft) below the finished grade within the NI footprint shall meet the criteria as stated in AP1000 DCD Section 2.5.4.5.3.

A 20 percent margin was provided in the design of the basemat, which was intended to account for possible soil property variations beneath the basemat at a site that may not meet the criteria for uniform soil sites. Additional analyses would be required for nonuniform soil sites. If the soil variations exceed the criteria as defined in AP1000 DCD Section 2.5.4.5.3, then the AP1000 DCD requires that an evaluation for nonuniform soil conditions be performed and this evaluation needs to be provided as part of the COL application. A procedure for evaluating the site-specific nonuniform soil condition is also provided in AP1000 DCD Section 2.5.4.5.3.

3.8.5.1.5.2 Soil Subgrade Modulus

In RAI-TR85-SEB1-05, the staff requested that the applicant provide a complete set of soil subgrade modulus values used for the AP1000 rock and soil cases. In a letter dated March 31, 2008, the applicant provided its response as follows:

- Subgrade moduli of 984.5, 502.7, 157.1, and 300.2 MPa/m^3 (6267, 3200, 1000, and 300 kips per cubic feet (kcf)) were used for HR, SR, SM and SS sites in the 2D [ ] parametric linear dynamic analyses described in Section 2.4.2 of TR-85. The results of the analyses for SR and SS were not used.

- Subgrade moduli of 984.5 MPa/m^3 (6267 kcf) and 157.1 MPa/m^3 (1000 kcf) were used for the HR and SM soil sites in the 2D [ ] nonlinear dynamic analyses described in Section 2.4.2 of TR-85.

- A subgrade modulus of 984.5 MPa/m^3 (6267 kcf) was used for HR in the 3D [ ] equivalent static nonlinear analysis for design of the basemat as described in Section 2.3.1 of TR-85.

- A subgrade modulus of 81.7 MPa/m^3 (520 kcf) was used for soil sites in the 3D [ ] equivalent static nonlinear analysis for design of the basemat as described in Section 2.6.1 of TR-85.

- A subgrade modulus of 40.8 MPa/m^3 (260 kcf) was used in the 3D [ ] equivalent static nonlinear parametric analysis for evaluation of the effect of a lower subgrade modulus as described in Section 2.7.1.1 of TR-85.

TR-85, Revision 1, indicates that the design of the NI basemat is based on the soil subgrade modulus corresponding to 81.7 MPa/m^3 (520 kcf) (comparable to the SM soil condition). This value of soil subgrade modulus was determined to be the governing soil case for design of the basemat considering the range of soil properties from HR to SS. To address soil conditions potentially softer than 81.7 MPa/m^3 (520 kcf), a study was performed to evaluate the effects of using lower stiffness values for the soil. Based on the applicant's March 31, 2008, and January 9, 2009, letters, the staff identified a number of items that still needed to be addressed regarding the evaluation for the appropriate range of subgrade modulus values. One of the concerns was that at other similar soil sites, subgrade modulus values as low as 6.3 MPa/m^3 (40 kcf) (static case) and about 12.6 MPa/m^3 (80 kcf) (dynamic case) have been identified. Therefore, in a follow-up to RAI-TR85-SEB1-05, the applicant was requested to explain whether the use of such low values had been considered and, if not, to provide the technical basis for not considering these values.

In a letter dated August 4, 2009, the applicant described the results of a study that was performed for a low soil modulus value of 12.6 MPa/m^3 (80 kcf) whose results were compared to the analysis using 81.7 MPa/m^3 (520 kcf) and 40.8 MPa/m^3 (260 kcf) soil moduli. To address the concern related to the design of the foundation, the RAI response indicates that a comparison of the 2D [ ] analysis results for all soil cases (FR, SR, UBSM, SM, and SS) was made to the soil case corresponding to a subgrade modulus of 12.6 MPa/m^3 (80 kcf). The results show that the soil bearing pressures for the 12.6 MPa/m^3 (80 kcf) soil case are very close to the 40.8 MPa/m^3 (260 kcf) (SS) case and they are bounded by the results for the 81.7 MPa/m^3 (520 kcf) case, which was used in the design of the basemat. The bending moments for the shield building at the base using the 81.7 MPa/m^3 (520 kcf) soil case bound the moments for the 12.6 MPa/m^3 (80 kcf) soil case. Therefore, the applicant concluded that these results demonstrate that the design of the foundation using a soil modulus value of 81.7 MPa/m^3 (520 kcf) is valid for soil subgrade moduli as low as 12.6 MPa/m^3 (80 kcf). For the soil bearing pressure demand, the comparisons presented in the RAI response show that the soil bearing pressure demand, used as interface criterion in the AP1000 DCD Tier 1, is acceptable since it bounds the soil bearing pressure for the 12.6 MPa/m^3 (80 kcf) case.

The staff found that the 2D [ ] analysis results demonstrate that the building responses for the 12.6 MPa/m^3 (80 kcf) soil modulus are bounded by the results for the 81.7 MPa/m^3 (520 kcf) soil case, which was used for design of the structures and for determining the soil bearing pressure demand. Also, for stability evaluation, the results presented in TR-85, Revision 1, show that the seismic shear force and overturning moment are lower when softer soil conditions are considered. Therefore, the stability evaluation performed by the applicant would also bound the results obtained with a reduced soil modulus of 12.6 MPa/m^3 (80 kcf). Based on the above discussion, the staff concludes that the soil cases used by the applicant for design, soil bearing pressure demand, and stability evaluation address the staff's concerns regarding subgrade moduli values lower than 81.7 MPa/m^3 (520 kcf). Therefore, RAI-TR85-SEB1-05 is resolved.

3.8.5.1.5.3 Assumption of Uniform Soil Pressure beneath the Basemat

The applicant assumed uniform soil pressure acting on the bottom of the basemat in its analysis for bending moments and shear forces in the basemat. It is a well-known phenomenon in soil mechanics that the soil pressure is higher at the edge of the basemat than it is away from the edge, which is referred to as the Boussinesq effect. Therefore, in RAI-TR85-SEB1-32, the staff requested that the applicant demonstrate that the use of the uniform soil springs for the design of the basemat is justifiable, where the actual distribution of the soil stiffness would not be uniform.

The RAI responses, dated June 23, 2009, and October 19, 2009, presented the results of a study that compared soil bearing pressures due to dead load at the bottom of the basemat from the uniform soil springs and the finite element representation of the soil. However, these results showed that the soil bearing pressure along the horizontal interface between the basemat and the soil do not appear to compare well in some regions. Furthermore, separate moment contour plots were provided for the basemat corresponding to each soil stiffness representation; however, without a direct quantitative comparison of member forces it is difficult to judge that the use of the uniform soil springs for the design of the foundation is acceptable. In a follow-up RAI, the staff requested that the applicant clearly demonstrate that the bending moments and shear forces in the basemat using uniform soil springs are acceptable by providing quantitative data from the study at locations in the basemat that govern the design.

Based on the applicant's letter dated June 19, 2010, a study was performed to compare the uniform soil spring approach with the more accurate finite element soil representation that is able to capture the Boussinesq effect in soils. This study showed that the soil pressures are not uniform and that some member forces in the critical sections in the basemat were larger using the finite element soil model. The applicant tried to scale the prior design results to show that the design is still adequate for the increased loads. However, the response to the RAI did not adequately demonstrate that the design met the code limits.

In a letter dated July 30, 2010, the applicant provided the re-evaluation for the basemat design using the increased loads from the finite element model for the critical (governing) sections and using the permissible redistribution of moments in accordance with the ACI 349 Code. In addition, the applicant provided the results for the various 100-40-40 seismic combination methods used for the design of the basemat. The staff's review of the response determined that several items still needed to be addressed, primarily because the response to the RAI still did not adequately demonstrate that the design met the code limits. Nor was the use of the Westinghouse 100-40-40 method appropriate. Therefore, in a follow-up RAI, the staff requested that the applicant justify the use of the 20 percent moment redistribution; show that the reinforcement design meets code requirements; provide the comparison for the Westinghouse 100-40-40 method versus the ASCE 4-98 industry method; and demonstrate that there are no significant increases in the basemat forces due to potential concrete cracking.

In response to the above requests, the applicant's letter dated September 8, 2010, provided detailed information justifying the use of the 20 percent moment redistribution in accordance with the ACI 349 Code. In addition, according to the letter, a new study was performed to compare the results from a 2D nonlinear (with lift-off capability) equivalent static analysis using the Westinghouse 100-40-40 method with those from a 2D nonlinear (with lift-off) time history analysis. The study shows that the maximum basemat bearing pressure from the 2D static analysis with the Westinghouse 100-40-40 method in two dimensions is about 30 percent higher (i.e., more conservative) than that of the bearing pressure from the more accurate 2D dynamic time history analysis approach. To address the effect of concrete cracking on the basemat forces, the applicant performed another study, which provided a comparison of the FRS at representative locations in the NI, which shows that the ZPAs obtained from the nonlinear analysis (that considers cracking of concrete) were reasonably close to the ZPAs obtained from the linear analysis using a stiffness reduction factor of 0.80, which was assumed in the design basis analysis.

The staff review of the response concluded that: (1) the justification for the use of 20 percent moment redistribution is acceptable because the information provided demonstrates that the provisions in ACI 349 regarding negative moment redistribution have been satisfied; (2) the basemat design based on the 2D nonlinear (with lift-off) equivalent static analysis using the Westinghouse 100-40-40 method is conservative based on the applicant's study comparing the results to the more accurate 2D nonlinear time history analysis, which inherently includes the phasing of the different input components; and (3) there is no significant increase in the basemat forces due to concrete cracking in the NI, because another study was performed to demonstrate that the use of the 0.8 stiffness reduction factor adequately accounts for cracking. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and TR-85, which resolve this issue.

3.8.5.1.5.4 Load Combinations and Reinforcement Design

As a result of the staff's review of TR-85, a number of questions were identified related to the load combinations and design of the basemat reinforcement. These questions were captured in RAI numbers TR85-SEB1-28, TR85-SEB1-29, and TR85-SEB1-30. As a result of these RAIs, the applicant made a number of revisions in the analyses and design methods to address these issues. The description provided below presents the staff's evaluation of the key issues related to the load combinations and design of the basemat reinforcement.

In RAI-TR85-SEB1-28, the staff requested that the applicant explain why the load combinations presented in the TR-85 were not consistent with those in Table 3.8.4-2 of the AP1000 DCD. In a letter dated December 2, 2008, the applicant provided a mark-up of AP1000 DCD Table 3.8.4-2 to be consistent with the revised TR-85. The staff finds that the new load combinations in the mark-up of AP1000 DCD Table 3.8.4-2 and in the revised TR-85 are in accordance with the ACI 349-01 Code, and, thus, are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD table, which resolves this issue.

In RAI-TR85-SEB1-29, the staff requested that the applicant describe the design approach used for the basemat in accordance with ACI 349-01. The staff also asked whether every 3D [ ] finite element is designed for the resultant forces in accordance with the ACI 349 Code and whether this process is automated by using a computer code or by hand calculations. In a letter dated October 19, 2007, the applicant stated that the design procedure is described in [ ], Revision 1, Section 4.2, "Calculation Approach/Methodology," and the calculation process is automated by a computer code. During the review of the shield building design, the staff found a potential error in the code. In the applicant's letter dated July 9, 2010, the response provided an explanation as to why some of the results from the computer code may have appeared as an error but they were not. The RAI response explained that the negative value of shear shown in the computer code results indicates that the code has detected that the concrete is in tension beyond its limit. The computer code does not use the strength provided by the concrete in that case. Based on the review of the design approach presented by the applicant for the basemat, the use of the ACI 349-01 Code for sizing the concrete sections and selection of reinforcement, and the information provided in the RAI response, which explained why negative values for shear may appear in the results generated from the computer code, the staff concludes that the design approach is acceptable. Therefore, RAI-TR85-SEB1-29 is resolved.

3.8.5.1.5.5 Minimum Required Soil Friction Angle, Settlement Criteria for the NI Structure, and Construction Sequence

Section 5.1 of TR-85 presents the proposed revisions to AP1000 DCD Tier 2, Table 2-1, which includes the site parameters including those for the soil media. Section 5.2 presents the proposed revisions to AP1000 DCD Tier 1, Table 5.0-1, which also includes the site parameters for the soil. Considering that the foundation of the AP1000 design has been extended to soil sites, in RAI-TR85-SEB1-36, the staff requested that the applicant include, in both tables, two additional parameters, which are needed for the structural design of the NI: a minimum required soil friction angle of 35 degrees beneath the basemat and settlement criteria for the NI structure.

In a letter dated March 31, 2008, the applicant provided the following response:

a) The minimum required soil friction angle of 35 degrees has been added to both Tables 2-1 and 5.0-1.

b) AP1000 DCD Section 2.5.4.6.11 requires the COL applicant to evaluate settlement at soil sites. The effect of settlement on the NI basemat during construction has been considered in the design of the NI as described in Section 2.5 of the report and in AP1000 DCD Section 3.8.5.4.2. These analyses considered the flexibility of the basemat during construction by performing a nonlinear analysis of the soil and NI. The nonlinear analyses are described in the applicant’s response to RAI-TR85-SEB1-19, dated March 31, 2008. The analyses used the NI05 building model described in AP1000 DCD Appendix 3G. The analyses considered an SS site with properties selected to maximize the settlement during construction. Immediate settlements were based on elastic properties of the foundation medium, while the time-related settlements used creep parameters established by comparison against one-dimensional consolidation theory. These analyses show total settlements of about one foot.

The applicant has established guidance on settlement for the COL applicant in the RAI response. The acceptable criteria are as follows: Acceptable differential settlement between buildings without additional evaluation is identified as 7.6 cm (3 in) between the NI and the Turbine Building, the Annex Building, and the Radwaste Building. The 7.6 cm (3 in) is measured from the center of the Containment Building to the center of the Turbine Building, center of the Annex Building, or the center of the Radwaste Building. Each building, including the NI, also has a settlement criterion of no more than 1.3 cm (½ in) in 15.2 m (50 ft) in any direction. The NI also has an acceptable maximum absolute settlement value of 7.6 cm (3 in). If site-specific settlement analyses predict settlements below the values in this table, the site is acceptable without additional evaluation. If the analyses predict greater settlement, additional evaluation will be performed. This may include specification of the initial building elevations, specification of the stage of construction and settlement for making connections of systems between buildings, etc. It would also include review of the effect of the rotation of buildings and its effect on the gap between adjacent structures. These analyses would provide the basis for review of settlement measurements during construction and subsequent operation.

Regarding part a) of the RAI response, the staff noted that in a letter dated June 10, 2009, the applicant indicated that a soil internal friction angle of 35 degrees is required beneath the basemat and it is specified in Table 2-1 of the AP1000 DCD, and that the second paragraph of AP1000 DCD Section 2.5.4.6.2 is revised to state that if the minimum soil angle of internal friction is below 35 degrees, the COL applicant will evaluate the seismic stability against sliding as described in Section 3.8.5.5.3 using the site-specific soil properties. The applicant also decided to remove the criterion for the soil friction angle of 35 degrees from the prior versions of AP1000 DCD Tier 1, Table 5.0-1, “Site Parameters.” After reviewing the applicant’s submittals, the staff requested that the applicant address several issues discussed below.

During the August 10, 2009 audit, the staff informed the applicant that if a site-specific evaluation is required for sliding because the soil friction angle is less than 35 degrees, then Section 3.8.5.5.4 of the AP1000 DCD should also add the evaluation requirement for overturning stability. In addition, the staff considered the demonstration of a site soil friction angle of 35 degrees to be a key site parameter in the stability evaluations and other analyses, such as determining the soil pressure loads for the design of the NI foundation walls. Therefore, this criterion should remain in AP1000 DCD Tier 1, Table 5.0-1. In a letter dated September 22, 2009, the applicant provided a proposed mark-up of AP1000 DCD Tier 1, Table 5.0-1, and AP1000 DCD Tier 2, Section 2.5.4.6.2, to incorporate the requirement for a site-specific evaluation when the soil friction angle is less than 35 degrees. However, the wording in AP1000 DCD Table 5.0-1, for the requirement of a site-specific evaluation, needs to be clarified so that it is clear that a stability evaluation should be performed for both sliding and overturning stability. In a letter dated May 14, 2010, the applicant revised the wording in the proposed mark-ups to AP1000 DCD Tier 1, Table 5.0-1, and AP1000 DCD Tier 2, Section 2.5.4.6.2. Since the AP1000 DCD markups specify the requirement for a minimum soil angle of internal friction of 35 degrees, and if it is less than 35 degrees, then the COL applicant will perform a site-specific analysis to demonstrate stability (sliding and overturning), the staff’s review of the information concluded that the response is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and table, which resolve this issue.

Regarding part b) of the RAI response, the staff observed that if acceptable soil sites are already known to cause potential settlements of as much as one foot as previous studies have indicated, then the construction settlements will in fact exceed the listed limitation of 7.6 cm (3 in) for most soil sites. The staff requested that the applicant explain: (a) what should be the detailed plan that the COL applicant needs to implement when the predicted settlements in fact exceed 7.6 cm (3 in); and (b) if any of the predicted settlements are less than 7.6 cm (3 in) for the total settlement, as well as less than the other acceptance values presented in AP1000 DCD Table 2.5-1, while the actual measured settlements during construction are found to exceed these values before completion of construction, what is the impact on the ongoing construction process and what the COL applicant is supposed to do at that time.

In the applicant’s letters dated December 2, 2008, and July 21, 2009, additional information was provided and one of the settlement threshold values was revised. The limit of acceptable settlement without additional evaluation was raised to 15.2 cm (6 in) for the total NI foundation mat. The RAI response also explained what steps would be taken in case the COL applicant’s predicted settlement analysis for the site-specific conditions exceeds these limits.

The staff reviewed the information regarding the settlement criteria and concluded that the applicant has evaluated the effects of settlement on the structural integrity of the NI and that conservative settlement threshold values (i.e., lower than the settlement values used for evaluation of the NI) have been proposed for inclusion in the AP1000 DCD. However, as requested in the original RAI and supplemental RAIs, the settlement criteria in the proposed mark-up of AP1000 DCD Tier 2, Table 2.5-1, should also be presented in AP1000 DCD Tier 1, Table 5.0-1.

In response to the above request, the applicant's letter dated June 21, 2010, indicated that the settlement criteria in the proposed mark-up of AP1000 DCD Tier 2, Table 2.5-1, are added to AP1000 DCD Tier 1, Table 5.0-1. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

In Section 2.5 of TR-85, the first paragraph states that in the expected basemat construction sequence, concrete for the mat is placed in a single placement. The last sentence of the same paragraph states that once the shield building and auxiliary building walls are completed to El. 25.1 m (82 ft-6 in), the load path changes and loads are resisted by the basemat stiffened by the shear walls. In RAI-TR85-SEB1-17, the staff requested that the applicant address several items related to the construction sequence. The applicant was requested to address issues related to the concrete pour of such a massive single concrete placement, how residual stresses at the junction between the shear walls and the shield building are calculated considering the construction sequence, and where in the AP1000 DCD the requirement to follow the construction sequences considered by the applicant in the design of the NI structures is located.

In a letter dated March 31, 2008, the applicant provided information to address the various items identified in the RAI. Regarding the construction sequence, the applicant described three construction sequences that were evaluated for an SS site to demonstrate construction flexibility within broad limits. The acceptability of the construction sequence used by the COL applicant is addressed by an ITAAC. The three construction sequences are as follows:

- A base construction sequence, which assumes no unscheduled delays.
- A delayed shield building case, which assumes a delay in the placement of concrete in the shield building while construction continues in the auxiliary building.
- A delayed auxiliary building case, which assumes a delay in the construction of the auxiliary building while concrete placement for the shield building continues.

The applicant indicated that analyses of alternate construction scenarios showed that member forces in the basemat are acceptable subject to the following limits imposed for SS sites on the relative level of construction of the buildings prior to completion of both buildings at El. 25.1 m (82 ft 6 in):

- Concrete may not be placed above El. 25.6 m (84 ft 0 in) for the shield building or CIS.
- Concrete may not be placed above El. 35.8 m (117 ft 6 in) in the auxiliary building, except in the CA20 structural module where it may be placed to El. 41.1 m (135 ft 3 in).

Based on the staff's evaluation of this response and follow-up RAI responses, the applicant was requested to revise the RAI response and Sections 2.5 and 3.8.5 of the AP1000 DCD to clearly state that in addition to satisfying settlement criteria the construction sequence limitations presented in Section 3.8.5.4.2 must be satisfied by the COL applicant. In the letter dated October 19, 2009, the applicant provided the proposed mark-up of AP1000 DCD Sections 2.5 and 3.8.5.4.2. The proposed wording indicates that the construction sequence limitations are only applicable to soil sites and not foundations identified by the applicant as SR, FR, or HR. The staff requested that the applicant justify why no construction sequence limitations are needed for the stiffer foundation materials.

In the applicant's letter dated July 15, 2010, the response to RAI-TR85-SEB1-17 indicated that the construction of the AP1000 will satisfy the construction sequence limits shown in AP1000 DCD Section 3.8.5.4.2 or a site-specific analysis of settlement and member forces will be completed. These limits do not apply to AP1000 units with a soil profile where Vs exceeds 2286.0 m/s (7500 fps). The Vs at the bottom of the basemat (i.e., locally) can drop to 2286.0 m/s (7,500 fps), while maintaining a Vs equal to or above 2438.4 m/s (8,000 fps) at the lower depths. The staff reviewed the proposed mark-ups to the AP1000 DCD and concluded that they are acceptable because: (1) the AP1000 was designed for the various construction sequences; and (2) the construction sequence limitations used in the SS evaluation are imposed on all soil conditions except for rock conditions having a Vs greater than 2286.0 m/s (7,500 fps). In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

3.8.5.1.5.6 The Effect of Ground Water on Nuclear Island Structures

The design of the AP1000 plant is based on saturated soil conditions. In RAI-TR85-SEB1-40, the staff requested that the applicant explain whether unsaturated conditions were also considered in performing any SSI analyses to determine the effects of unsaturated soils on the response of the NI in terms of member forces, deformations, and FRS.

In a letter dated May 27, 2009, the applicant indicated that it performed a time history analysis using a saturated and unsaturated SM soil profile (Poisson's ratio = 0.35) and compared the FRS of the two analyses. Generic SSI analyses for the AP1000 assume the water table to be at grade level with saturated soil properties supporting the NI. The unsaturated soil profile was produced from a SHAKE analysis where the water table was assumed to be well below the NI. The results of this analysis indicated that the depth of the water table used for SSI analyses has a negligible effect on the FRS at the key nodes. This study shows that generally the FRS for these two cases are very close to one another, with the spectra from saturated conditions somewhat higher in a few isolated cases. Since the FRS differences between the two models are negligible, no additional analyses are required to compare member forces or deformations.

The staff reviewed the applicant's submittal regarding the effect of saturated and unsaturated soil conditions on NI structures, and found the applicant's approach to address the issue reasonable and acceptable. Since the study shows that, generally, the FRS for both saturated and unsaturated cases are very close to each other, with the spectra from saturated conditions somewhat higher in a few isolated cases, and the design of the AP1000 plant is based on the saturated conditions, the staff concludes that the AP1000 design using saturated soil conditions adequate and acceptable. Therefore, RAI-TR85-SEB1-40 is resolved.

3.8.5.1.5.7 Potential Uplift/Sliding between CIS and Containment, and between Containment and Basemat

In RAI-TR85-SEB1-12, the staff requested that the applicant explain how the potential uplift and sliding between the CISs concrete base and the steel containment shell is addressed for the various soil conditions, and provide the basis for the statement in Section 3.8.2.1.2 of the AP1000 DCD, which indicates that the shear studs provided between the containment and concrete basemat below the containment are not required for design basis loads, but provide additional margin for earthquakes beyond the SSE.

In a letter dated October 19, 2007, the applicant stated that its analyses of stability for the HR site demonstrated that there was no uplift or sliding at the interface of the CIS and the CV. These analyses showed potential uplift of the CV and CISs from the NI basemat for the RLE. Based on these analyses, the applicant provided shear studs between the CV and the NI basemat to provide additional margin for the RLE. These studs were then designed to accommodate pressurization of the CV. The number of studs required for containment pressure was more than double the number required for seismic overturning for the RLE at the HR site. Revision 1 of TR-85 describes the analysis, which demonstrated that no uplift or sliding occurs between the CIS and the containment, and between the containment and the basemat for both design basis SSE level of 0.3g and RLE level of 0.5g PGA for HR and all soil conditions. Based on this, RAI-TR85-SEB1-12 is resolved.

3.8.5.1.5.8 The 100-40-40 Method for Combining Three Components of Earthquake Motions

AP1000 DCD Section 3.7.2 states that the 100-40-40 method is used for combining the three components of earthquake motions for the NI basemat analyses, CV analyses and shield building roof analyses. NRC regulatory guidance in RG 1.92 and NUREG-0800 Section 3.7.2 indicates that the use of the 100-40-40 combination method is only acceptable for combining the co-directional responses, such as Mxx due to north-south, east-west, and vertical directions in order to obtain a combined Mxx. However, it does not appear from a review of TR-85 and AP1000 DCD Section 3.8 that the applicant has combined the three components in accordance with RG 1.92 and industry standard ASCE 4-98. This issue was also identified during the staff's evaluation of TR-57 and APP-1200-S3R-003 for the shield building, which is discussed in Section 3.8.4.1.1 of this report. The issue of the proper implementation of the 100-40-40 method was captured under RAI-TR85-SEB1-27.

As indicated in a letter dated July 3, 2010, the applicant's approach for the 100-40-40 method (Westinghouse 100-40-40 method) was used for both seismic linear and nonlinear equivalent static analyses for the design of the NI basemat, the SCV and the shield building roof. In addition, the applicant also indicated that: (1) for the basemat, the justification for using the applicant's 100-40-40 method was addressed under RAI-TR85-SEB1-32; (2) for the SCV, the adequacy of using the applicant's 100-40-40 method for the SSE loading condition was confirmed by a direct comparison of the combined seismic stress results against those from the more accurate time history analysis; and (3) for the shield building roof, a comparison of the applicant's 100-40-40 method to the ASCE 4-98 method was made. For the shield building roof analysis and design, the applicant developed equivalent static accelerations, such that the resulting member forces would envelope those from the RSA, performed for the input motion applied at the foundation level enveloping all the soil cases. The justification for using the applicant's 100-40-40 method was provided by comparing the combined member forces corresponding to the 24 cases of the applicant's 100-40-40 method with the member forces from the ASCE 4-98 method.

The staff's review of the information provided to the staff concluded that: (1) the justification for using the applicant's 100-40-40 method under RAI-TR85-SEB1-32 is acceptable since this approach is coupled directly with the basemat design issue under RAI-TR85-SEB1-32, which was previously reviewed above; and (2) the response for the SCV is acceptable, because the results provided show that the applicant's 100-40-40 method produced conservative results when compared with the more accurate time history analysis results. However, the response for the shield building roof provided insufficient information, primarily because the comparison of the applicant's 100-40-40 method with the ASCE 4-98 method is only made for member forces and not the final design parameter (e.g., required reinforcement for concrete members or stress level for steel members). Therefore, it is not clear that the applicant's 100-40-40 method is adequate. To address the issue of the proper implementation of the 100-40-40 method for the shield building roof design, the staff requested that the applicant identify the locations where the 100-40-40 method was applied in the shield building roof design; determine the maximum required reinforcement (or stress levels for steel members) using the 24 cases of the applicant's 100-40-40 method (as is done in the applicant's design process) and compare these results with the required reinforcement (or stress levels for steel members) using the NRC-accepted SRSS method or the ASCE 4-98 100-40-40 method.

In response to the above requests, the applicant's letter dated September 23, 2010, identified that the air inlet, the tension ring and the composite radial steel beams were designed using the applicant's 100-40-40 method, and provided figures and descriptions of the models used for the design of the shield building roof. To justify the use of the applicant's 100-40-40 method, the applicant presented comparisons for the final design parameters for these members showing that, although in some cases the applicant's 100-40-40 method was nonconservative when compared with the SRSS method or the ASCE 4-98 method; in all cases the design of these members is still acceptable. This was demonstrated for concrete members by showing that the required reinforcement using the NRC-accepted SRSS method was less than the provided reinforcement and for steel members by showing that the calculated stresses using the NRC-accepted SRSS method were less than the code allowable.

In a subsequent revision to the AP1000 DCD and TR-85, the applicant made appropriate changes to the DCD and TR-85 text, which resolve this issue.

3.8.5.1.6 Record Keeping Issues

Sections 2.3.1, 2.4.1, 2.4.2, and 2.6.1 of TR-85 indicate that equivalent static nonlinear analysis, 2D [ ] analysis, 2D [ ] linear dynamic analysis, 2D [ ] nonlinear time history analysis, 3D [ ] equivalent static nonlinear analysis, and others were performed. In RAI-TR85-SEB1-04, the staff requested that the applicant develop a table (or tables) similar to AP1000 DCD Tables 3.7.2-14 and 3.7.2-16 to show: (1) the purpose of each analysis; (2) the model type(s); (3) analysis method(s); (4) soil condition(s); (5) loads, load combinations, combination method (for combining loads and directional combinations for SSE); (6) governing design loads; and (7) reference location in TR-85 or other reports for the detailed description.

In a letter dated December 4, 2007, the applicant provided revisions to the AP1000 DCD tables to show the additional information requested in this RAI and to reflect the changes in the methodology described in other RAI responses. Although sufficient information to describe the evaluations performed for the bearing pressure demand, foundation stability, and design of the basemat, has been provided in this and other RAI responses and in TR-85, Revision 1, the staff could not identify where a description of the evaluations for bearing pressure demand and foundation stability are presented in the AP1000 DCD. Therefore, the staff requested that the applicant include in the AP1000 DCD a description of the evaluations performed for the bearing pressure demand and foundation stability, which consists of a summary of the analyses presented in TR-85, Revision 1.

In a letter dated June 4, 2009, the applicant provided the proposed changes to the AP1000 DCD that describe in more detail the soil bearing pressure evaluation in TR-85. This information will be added to Appendix 3G of the AP1000 DCD. In addition, the applicant indicated that the changes to the AP1000 DCD related to the stability evaluation are given in a revision to RAI-TR85-SEB1-10, along with a summary of the 2D nonlinear sliding evaluation. Thus, the description of the stability evaluation for inclusion in the AP1000 DCD is evaluated separately under the staff's assessment of RAI-TR85-SEB1-10 in this SER. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

TR-85 is referenced in AP1000 DCD Section 3.8.5 and it includes key analysis and design information of the foundation. TR-09 is referenced in AP1000 DCD Section 3.8.2.4.1 and it includes key analysis and design information for the containment. TR-57 is referenced in Revision 17 to the AP1000 DCD Section 3.8.4 and it includes key analysis and design information for the CIS, auxiliary, and the shield building critical sections. The staff notes that the applicant clarified the design basis by letters dated October 21, 2010, whereby they withdrew TR-57 and provided mark-ups of the DCD to show the removal of references to TR-57 and stated where the information, as updated, appears in the proposed DCD and an appendix thereto. APP-1200-S3R-003 is referenced in AP1000 DCD Section 3.8.4 and it describes key analysis and design information for the shield building. Any revisions to the Tier 2* information will be subject to the NRC review and approval to avoid unintended safety consequences. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and table, which resolve this issue.

In RAI-TR85-SEB1-39, the staff requested that the applicant identify the specific design reports, calculations, and reports related to various studies that are applicable to the analysis and design of the AP1000 NI basemat and foundation.

In a letter dated October 19, 2007, the applicant stated that:

> APP-1010-S3R-001, "AP1000 Design Summary Report: Nuclear Island Basemat," provides a detailed summary of the design of the NI basemat. It satisfies the guidelines of NUREG-0800 Section 3.8.4 and is available for NRC review during the structural audit.

The design summary report identifies the applicant's specific design reports, calculations, and reports applicable to the analysis and design of the AP1000 NI basemat and foundation. Some of the documents referenced therein are listed below. The criteria and methodology documents were previously reviewed during the audit of the basemat design on HR.

1. APP-GW-C1-001, "AP1000 Civil/Structural Design Criteria," Revision 1

2. APP-GW-S1-008, "Design Guide for Reinforcement in Walls and Floor Slabs," Revision 1

3. APP-GW-S1-009, “Design Guide for Thermal Effects on Concrete Structures,” Revision 0

4. APP-1000-CCC-001, “Verification of Design Macro for Reinforced Concrete Walls and Floors,” Revision 2

5. APP-1000-CCC-002, “Guidance on Checking Results of Design Macro Calculation,” Revision 0

6. APP-1010-S2C-003, “Macro to Calculate Required Reinforcement in Solid Elements,” Revision 0

7. APP-1010-S2C-004, “Basemat Liftoff, and CV Pressure Analyses for Nuclear Island with Soil,” Revision 0

8. APP-1010-CCC-001, “AP1000 Basemat Design Report,” Revision 2

9. APP-1010-CCC-003, “Basemat Design Studies, Effect of Soil Modeling,” Revision 0

10. APP-1010-CCC-004, “Basemat Design, Below Auxiliary Building,” Revision 1

11. APP-1010-CCC-005, “Basemat Design, Below Shield Building,” Revision 0

12. APP-1200-S2C-002, “ASB Thermal and Earth Pressure Analyses,” Revision 1

13. APP-1200-S2C-003, “Auxiliary Building Load Combinations and Loads for Finite Element Analyses,” Revision 0

14. APP-1000-CCC-005, “N.I. - Design Loads, Exterior Walls Below Grade,” Revision 1

15. APP-1000-CCC-004, “Nuclear Island Stability Evaluation,” Revision 1

16. APP-1000-S2C-064, “Effects of Basemat Liftoff on Seismic Response,“ Revision 4

17. APP-1000-S2C-065, “Nuclear Island Stick Model Analyses at Soil Sites,“ Revision 0

In an e-mail dated April 30, 2009, the applicant updated the documents related to the basemat design that are available for review. In the audit conducted during the week of May 4, 2009, the staff reviewed a number of these documents to ensure that the evaluations were performed in accordance with the AP1000 DCD and NRC regulatory guidance. The staff concluded that the applicant had identified the design reports, calculations, and reports related to the AP1000 NI basemat and foundation, and the staff had an opportunity to review some of these documents for technical adequacy. Therefore, RAI-TR85-SEB1-39 is resolved.

3.8.5.1.7 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the applicant’s application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 foundations as documented in AP1000 DCD, Revision 19, against the relevant acceptance criteria as listed above and in NUREG-0800, Section 3.8.5.

In subsequent revisions to TR-85, the applicant made appropriate changes to the report. Based on the review of these changes, staff concludes that APP-GW-GLR-044, TR-85, "Nuclear Island Basemat and Foundation," Revision 3, is acceptable because the analyses and design were performed in accordance with the ACI 349 Code, applicable RGs, and NUREG-0800, Section 3.8.5.

Therefore, the staff concludes that the design of the AP1000 foundations will continue to meet all applicable acceptance criteria. In summary, based on the above discussions, the staff finds that the design of the AP1000 foundation is acceptable.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each COL applicant would have to address these issues individually.

3.8.6 Combined License Information

Section 3.8.6, "Combined License Information" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revisions 16 and 17, the applicant made the following changes to Section 3.8.6 of the certified design:

1. In DCD Revision 16, the applicant revised Section 3.8.6.1, Containment Vessel Design Adjacent to Large Penetrations. This revision eliminated this COL information item because the applicant indicated that the information had been addressed in APP-GW-GLR-005 (TR-09) and the applicable changes were incorporated into the DCD.

2. In DCD Revision 16, the applicant also revised Sections 3.8.6.2 through 3.8.6.4, to delete the remaining COL information items relating to the PCS water storage tank examination, as-built summary report, and in-service inspection of containment vessel. No explanation for this deletion was provided in DCD Section 3.8.

The staff evaluation of the changes to the COL information item in AP1000 DCD Section 3.8.6.1 related to the CV design adjacent to large penetrations is presented in Section 3.8.2.4.1 of this report, where the staff reviewed APP-GW-GLR-005, Revision 0 (TR-09). In subsequent revisions to the AP1000 DCD and TR-09, the applicant made appropriate changes to the DCD and report text, which resolve this issue.

The staff noted that the applicant removed the COL information items in AP1000 DCD Sections 3.8.6.2 through 3.8.6.4 that relate to the PCS water storage tank examination, as-built summary report, and the inservice inspection of containment vessel. Therefore, in RAI-SRP3.8.6-SEB1-01, the staff requested that the applicant restore these items in AP1000 DCD Section 3.8.6 which were discussed in the prior versions of AP1000 DCD Sections 3.8.1 through 3.8.5. In a letter dated February 19, 2009, the applicant indicated the following:

For the COL information item in AP1000 DCD, Section 3.8.6.2, the requirement to examine the PCCWST is redundant with Design Commitment 10, ITAAC Item ii of Tier 1, Table 3.3-6.

For the COL information item in AP1000 DCD Section 3.8.6.3 the requirement to prepare an as-built summary report is redundant with Design Commitment 2.a, ITAAC Item I of Tier 1 Table 3.3-6.

For the COL information item in AP1000 DCD Section 3.8.6.4, the inservice inspection of the containment is required by NRC regulations including 10 CFR 50.55a. There is also a commitment for inservice inspection of the containment in AP1000 DCD Section 6.6.1.

The staff's review of the information provided in the RAI response has led to the conclusion that the deletion of the COL information item in AP1000 DCD Section 3.8.6.3 is acceptable because the information is redundant with an ITAAC and, in the case of Section 3.8.6.4, is already required in 10 CFR 50.55a. However, in the case of the COL information item in AP1000 DCD Section 3.8.6.2, the ITAAC referred to by the applicant does not fully capture the examination requirements in AP1000 DCD Section 3.8.4.7 that the previous COL information item referred to. The ITAAC addresses examination for leakage and measurement of elevation at two locations before and after filling of the PCS storage tank. AP1000 DCD, Section 3.8.4.7, however, provides additional requirements for examination of excessive cracks in accordance with ACI 349.3R-96. Therefore, in a follow-up RAI, the applicant was requested to include this additional commitment as part of the subject ITAAC or provide the technical basis for excluding it.

In a letter dated September 9, 2009, the applicant agreed to revise the ITAAC in AP1000 DCD Tier 1, Table 3.3-6, to fully capture the examination requirements in AP1000 DCD Section 3.8.4.7 for the PCS storage tank. In addition, the applicant identified that a revision in AP1000 DCD Tier 2, Section 3.8.4.7, was required for testing to be performed to measure the leakage from the PCS storage tank based by measuring the water flow out of the leak chase collection system.

The staff's review of the applicant's September 9, 2009, response determined that the proposed revisions to ITAAC Table 3.3-6 and AP1000 DCD Section 3.8.4.7 are still not consistent. The commitment in AP1000 DCD Section 3.8.4.7 to inspect the PCS tank for significant cracking in accordance with ACI 349.3R-96 is not included in the ITAAC. In addition, the inspection identified in the ITAAC is applicable to the PCS tank boundary and the shield building tension ring while in the case of AP1000 DCD Section 3.8.4.7, the inspection is applicable to the PCS boundary and the shield building roof above the tension ring. The applicant needed to explain whether the inspection would be performed for all three structural regions (PCS tank boundary, shield building roof, and tension ring) and revise both sections of the AP1000 DCD to be consistent. In a follow-up RAI, the staff requested that the applicant address both items discussed above.

In response to the above requests, the applicant's letter dated June 18, 2010, explained that the references to specific standards, such as ACI 349.3R-96, are not included in Tier 1 because this is an established practice in the preparation of Tier 1 information. Since ITAAC Table 3.3-6 in the AP1000 DCD, Revision 15, did not identify the ACI 349.3R-96 standard, but AP1000 DCD Section 3.8.4.7 did, the staff concludes that it is acceptable now to follow the same approach in the current AP1000 DCD.

To address the inconsistency between the proposed revisions to the ITAAC and the AP1000 DCD on the inspection regions, the applicant explained that the design now has steel plates as the outer surface of the tension ring for the enhanced shield building, and concrete cracking in the tension ring region will not be visible; therefore, Table 3.3-6 in the ITAAC will be revised to clarify that the inspection for visible excessive cracking will be performed for the roof above the tension ring and the PCS tank boundary. Since the proposed revisions to the ITAAC Table 3.3-6 and AP1000 DCD Section 3.8.4.7 are now consistent, the staff concludes that this part of the response is also acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text and table, which resolve this issue.

Shield Building COL Items

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and the DCD (up to and including Revision 15 of the AP1000 DCD) were acceptable and that the application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff concludes that if the items identified above are resolved, the COL information items will meet the applicable acceptance criteria, and that the proposed changes are properly documented in the updated AP1000 DCD. This is based on the additional evaluation report (TR-09) for the containment design adjacent to large penetrations, the inclusion of two ITAAC for the examination of the PCS water storage tank and the as-built summary report, and the existing requirements in 10 CFR 50.55a for the inservice inspection of the containment.

3.8.7 Conclusions

The NRC staff concludes that the proposed changes to the AP1000 DC, related to the design of Category I Structures, as described in the evaluation above, are acceptable because they satisfy the applicable requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 5, 16 and 50; 10 CFR 50.44; and 10 CFR 50.55(a).

Revision 19 to the AP1000 DCD provides sufficient information to satisfy the applicable requirements of the above regulations.

The changes to the DCD implementing the revised AP1000 design meet the standards of Criterion vii of 10 CFR 52.63(a)(1) in that they contribute to increased standardization; without these DCD changes each COL applicant would have to address these issues individually.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The evaluation is performed for AP1000 DCD, Revision 17. The applicant proposed editorial and minor technical changes and clarifications to the section including adding daily load follow operations to the Level A service conditions; redefining RCP startup and shutdown cases; and defining loading and unloading operations. In addition, the applicant proposed to add WESTEMS design computer code to AP1000 DCD Table 3.9-15 for application of the fatigue analysis of components.

3.9.1.1 Evaluation

AP1000 DCD Tier 2 Section 3.9.1.1.1.4 addresses the unit loading and unloading operations associated with power changes of 5 percent per minute between 15 percent and 100 percent power levels. The number of loading and unloading operations is defined as 2,000 each for the 60-year plant design. RAI-SRP3.9.1-EMB1-01 requested that the applicant provide the technical basis for splitting the 2,000 occurrences from the original 19,800 occurrences for the plant loading and unloading at 5 percent of the full power per minute for the normal plant startup/shutdown, and loading resulting from all service Levels B, C, and D transients that result in a reactor trip.

In its September 5, 2008, response to RAI-SRP3.9.1-EMB1-01, the applicant indicated that when the design transients for the AP1000 were initially established, it was decided to use the unit loading and unloading transient to cover the load follow and increase the number of these transients to cover a daily load follow. It is noted that this was a conservative approach since the load follow transient is less severe than the unit loading and unloading transient. As such, the daily load follow transient will be appropriately addressed rather than assuming the unit loading and unloading transient for most of the load follow requirement. The applicant used 2,000 occurrences of unit loading and unloading each to account for shutdowns and the recovery from service Level B, C, and D transients. The applicant noted that the 2,000 occurrences will cover the approximately 700 total service level B, C, and D transients and 1 (one) per month for loading and unloading each for 60 years. The applicant also noted that this frequency is larger than that at currently operating units and is considered bounding. The staff concurs with the applicant on the basis of its operating experience and concludes that use of 2,000 occurrences of unit loading and unloading is conservative and acceptable. RAI-SRP3.9.1-EMB1-01 is, therefore, closed.

AP1000 DCD Tier 2 added a new Section 3.9.1.1.1.19, “Daily Load Follow Operations” to Revision 16 to account for the one load follow operation per day that was included as a portion of the plant loading and unloading events for the design transients. RAI-SRP3.9.1-EMB1-02 requested that the applicant provide the basis of how the 17,800 cycles were determined for the daily load follow operations during the plant design of 60 years which with a 90 percent availability factor could result in 19,800 occurrences, and to discuss the basis that the load follow event could not coincide with the plant loading and unloading transients while they might occur at the same time.

In its September 5, 2008, response to RAI-SRP3.9.1-EMB1-02, the applicant noted that the total of unit loading and unloading transients combined with the daily load follow transient is 19,800 transients for 60 years of plant operation based on one transient per day with 90 percent plant availability factor. With the case of reduced power or in a load following mode, the nuclear power plant typically runs on a weekly cycle not a daily cycle. As such, it is assumed that a unit unloading and a daily load follow event would not occur on the same day. With 2,000 occurrences (each) for unit loading and unloading transients, the remaining 17,800 occurrences are made up of the daily load follow transients. The staff agrees with the applicant’s determination to use 17,800 occurrences for a daily load follow transient considering 2,000 occurrences conservative for unit loading and unloading transient as this case is much more severe than the daily load follow transient. Therefore, RAI-SRP3.9.1-EMB1-02 is closed.

As a result of the onsite technical review on October 20, 2008, the staff found that the fatigue analyses for the design of AP1000 seismic Category I components and supports were performed using a computer program called WESTEMS, which is not discussed in the AP1000 DCD Section 3.9.1.2, “Computer Code Used in Analyses,” nor listed in Table 3.9-15, “Computer Programs for Seismic Category I Components.” In its response to the staff’s

RAI-SRP3.9.1-EMB1-03, the applicant indicated that the DCD will be revised to add the WESTEMS computer program to Table 3.9-15. It also stated that the WESTEMS computer program was not previously reviewed and approved by the staff. On May 26 to 28, 2009, the staff conducted an audit of WESTEMS at the applicant's headquarters in Monroeville, Pennsylvania. The audit was not completed because not all the documents requested were available at the time of the audit. The follow-up review was completed at the end of September 2009 in the applicant's Twinbrook office in Rockville, Maryland.

During the audit, the staff discussed with the applicant the theoretical background, formulation, validation methods, and benchmarking problems pertaining to WESTEMS. The discussions including, in part, the RAIs the staff presented to the applicant during the exit meeting are described in the following paragraphs.

The staff reviewed the WESTEMS basis documents and identified that the stress peak/valley selection option using the stress evaluated with algebraic summation of three orthogonal moment components requires justification. The staff noted that the algebraic summation of three orthogonal vectors is mathematically incorrect and physically meaningless. The staff requested that the applicant provide technical justification for this option in selecting peak and valley times for the fatigue evaluation. This concern was identified as Open Item OI-SRP3.9.1-EMB1-05.

The WESTEMS program provided an option to eliminate peak/valley points during calculation. The staff noted that the computer output should not be modified after executing the program. The staff requested that the applicant provide the configuration control and limitations of the program for this option. This concern was identified as Open Item OI-SRP3.9.1-EMB1-07.

The staff performed an onsite review to discuss/resolve the above mentioned open items. The staff's onsite review summary report, dated December 9, 2010, identified the WESTEMS deficiency.

By a letter dated September 29, 2010, the applicant requested to remove WESTEMS from the DCD markup that adds WESTEMS to Table 3.9-15 of the DCD. In this letter, the applicant stated that the DCD need not include the WESTEMS program because the analyses in question are identified as COL Information Item 3.9-7 in the DCD and are not within the scope of the design certification amendment. The applicant also stated that it would use an appropriate analytical tool for performing the aforementioned analyses and the COL applicant has responsibility to close out the COL Information Item. The staff agreed that the COL applicant is responsible to close out COL Information Item 3.9-7 and fatigue analysis is part of the piping analysis. However, the staff was concerned that this tool should be provided as part of the methodology in the DCD. The staff acknowledged that the methodology available in the DCD in Revision 15 was complete such that the fatigue analysis could be performed without an additional tool. Also, DCD Tier 2, Section 3.9.2.1, states that the COL applicant will implement the NRC benchmark program using AP1000 specific problems if a piping analysis program other than those for design certification (PIPESTRESS, GAPPIPE, WCAN, and ANSYS) is used. This statement is marked as Tier 2*. The staff notes that use of a computer code as an analytical tool, as stated above, would require departure from the DCD based on the closure of the COL Item in Section 3.9.8.6 of the application. The closure is discussed in Section 3.12.1.2 of this report. On the basis that the applicant would return to the previously certified methodology, which was complete, and that any computer code added in the future would require benchmarking, the staff finds this acceptable. Therefore, Open Items OI-SRP3.9.1-EMB1-05 and OI-SRP3.9.1-EMB1-07 are closed.

3.9.1.2 Conclusions

Based on the letter dated September 29, 2010, the staff concludes that the applicant's request to remove WESTEMS from the DCD markup that adds WESTEMS to Table 3.9-15 of the DCD results in no change to the DCD for this item. On the basis mentioned above, the staff determined that all the open items related to WESTEMS are closed. The staff will evaluate piping design fatigue analysis to ensure piping integrity for safety at the time of COL item closure. The staff concludes that the DC amendment for Section 3.9.1 is acceptable.

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

3.9.2.3 Preoperational Flow – Induced Vibration Analysis and Testing of Reactor Internals

3.9.2.3.1 Summary of Technical Information

In AP1000 DCD, Revision 17, Section 3.9.2, "Dynamic Testing and Analysis," the applicant proposed changes to reactor internals and analysis. These changes included: addition of a flow skirt to the reactor vessel lower head, addition of neutron panels, relocation of radial support keys and tapered periphery on lower core support plate (LCSP), downcomer excitations and related responses, reduction of core shroud brace thickness, and RCP induced loads.

3.9.2.3.2 Evaluation

Section 3.9.2 of NUREG-1793 describes the AP1000 reactor vessel internals conformance with RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," November 2006, and NUREG-0800 Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components." The first AP1000 reactor internals design is classified as a prototype, as defined in RG 1.20. However, as stated in WCAP-16716, "AP1000 Reactor Internals Design Changes," the applicant does not consider the AP1000 reactor vessel internals a first-of-a-kind or unique design. Several units that have operating experience collectively have similar reactor vessel internal design features and are referenced in support of the AP1000 reactor vessel internals design.

The original reference plant for the applicant's three-loop plant reactor internals flow-induced vibration is H. B. Robinson. The results of vibrations testing at H. B. Robinson are reported in WCAP-7765-AR, "Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance," November 1973. With the addition of neutron panels to the reactor vessel internals design, the applicable referenced plant test has changed from Paluel 1 (no reactor shielding) to Trojan 1 (similar to current neutron panel AP1000 configuration). The applicant believes, as stated in WCAP-16716, that the change in referenced plant tests will not impact the conclusions in WCAP-15949-P, "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," Revision 2, April 2007.

The vibration testing for 17x17 fuel internals and inverted hat upper internals is reported in WCAP-8766, "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," May 1976 and WCAP-8516-P, "UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations," March 1975. The vibration testing of three-loop XL type lower core support structure in DOEL 4 is reported in WCAP-10846, "Doel 4 Reactor Internals Flow-Induced Vibration Measurement

Program," March 1985. The vibration evaluations of upper and lower internals assemblies for a four-loop XL plant are reported in WCAP-10865, "South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment," February 1985. The vibration testing of the core shroud lower internals design is reported in CE Report 10487-ME-TE-240-03, "A Comprehensive Vibration Assessment Program for Yonggwang 4 Nuclear Generating Station, Final Evaluation of Pre-Core Hot Functional Measurement and Inspection Programs," August 22, 1995.

The results of the Doel 3 and Doel 4 reactor internals vibration test programs have been utilized to perform the vibration assessment of the AP1000 reactor internals. The measured responses from Doel 3 and Doel 4 have been adjusted to the higher AP1000 flow rate to support the determination of the expected upper internals and lower internals vibration levels, respectively. The velocity through the core is approximately the same as that of Doel 4.

The results of the Trojan 1 tests showed that the lower internals vibrations are lower with neutron panels than with a circular thermal shield as reported in WCAP-8766.

The staff reviewed the relevant documents as stated above and evaluated the impact of changes in the reactor internals on the vibration evaluations of upper and lower internals assemblies. In addition, the staff reviewed the basis of the applicant's contention in WCAP-16716 that there is no impact on the conclusions in the DCD.

3.9.2.3.2.1 Addition of Flow Skirt to the Reactor Vessel Lower Head

The results of the computational fluid dynamics (CFD) calculations using the existing structures in the lower plenum along with the LCSP flow hole geometry indicated that the core inlet flow distribution needed to be adjusted to create a more uniform core inlet flow distribution. The core inlet flow distribution was improved by the addition of a flow skirt to the lower plenum of the reactor vessel.

CFD analyses of numerous configurations of the hardware in the lower reactor vessel have been made with the objective of obtaining a core inlet flow distribution that meets specifications established by the applicant's fuel group. It has been determined that flow distributions that meet the requirements are obtained with a flow skirt. A flow skirt is a perforated cylinder in the lower reactor vessel head that is attached to the reactor vessel bottom head. The flow skirt is attached to the lower head of the reactor vessel at the plant site after measurements for machining of the core barrel clevises have been completed. The attachment consists of welds across eight tabs that rest on support lugs provided on the reactor vessel lower head.

There is a circumferential weld between the spherical bottom vessel head and the conical transition to the cylindrical portion of the reactor vessel. The weld is just above the top surface of the flow skirt support lugs. There is some radial clearance between the outside of the flow skirt and the inside surface of the reactor vessel at the circumferential weld location. Examination Category B-N-2 of Section XI, Subsection IWB-2500, provides requirements for the visual (VT-3) examination of "interior attachments beyond the beltline region" of the reactor vessel. Vertical access for a pole-mounted camera is possible around the full circumference of the flow skirt with partial blockage at the four lower radial support keys located on the cardinal axes. It has been judged that the flow skirt and attachment welds could be inspected using VT-3 examinations. If any relevant condition is detected, IWB-3122 (prior to service) or IWB-3142 (inservice) provides options for correcting the condition. The staff reviewed the impact of the welds in generating additional vorticity and turbulence in the lower plenum region.

Based on its review the staff determined that additional information is needed for the staff to complete its review. Several welded joints have been introduced as a result of the addition of the flow skirt, as stated earlier. In RAI-SRP3.9.2-EMB1-07, the staff requested that the applicant discuss the potential for generation of vortices in the region of the flow skirt due to the presence of these welded joints as well as the flow skirt itself and the potential adverse effects on the response of other internals components. The applicant was also requested to discuss any tests related to the evaluation of the flow skirt performance.

In its June 20, 2008, response, the applicant stated, “Any vortices in this region would be proportional in size to the minimum open dimension between the vessel and the flow skirt. This will be on the order of 0.955 cm (0.376 in). Any vortices generated will therefore be too small and of too high a frequency (frequency is proportional to velocity divided by vortex dimension) to be of concern. If anything, the flow skirt will tend to dissipate any larger vortices that may be produced by the flow around the radial keys. The fact that the flow skirt makes the lower plenum flow field more uniform is an additional benefit. Because of this, there is a diminished possibility of large velocity gradients entering the lower plenum from the vessel down comer. Lower velocity gradients (greater flow uniformity) also diminish the probability of large vortex-formation. Flow skirts of similar design have been successfully used in operating System-80 plants. A scale model flow test, which includes the flow skirt and its connections to the reactor vessel, is planned as a confirmatory test.”

Based on its review, the staff finds that the applicant has provided a reasonable and satisfactory explanation for a diminished likelihood of large vortex formation in the lower plenum region and Open Item OI-SRP3.9.2-EMB1-07 is closed.

3.9.2.3.2.2 Addition of Neutron Panels

To provide flexibility in the core design over the life of the plant, end-of-life reactor vessel fluence calculations were made assuming a radial core power distribution of higher power fuel assemblies in the outmost peripheral locations than in a normal low leakage core. To maintain the end-of-life reactor vessel fluence values at less than the maximum allowed in RG 1.99, neutron panels were attached to the outside diameter of the core barrel. The resulting reactor vessel fluence is 8.9E19 n/cm2 (E > 1.0 MeV) at the end of the 60-year life. Neutron panels have been used on the recent Westinghouse reactor internals designs. They reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values and provide a relatively rigid structure that has a smaller downcomer cross-sectional area than a full cylinder.

The neutron panels are located at four circumferential locations where fuel assemblies are closest to the reactor vessel (0, 90, 180, and 270 degrees). Each pad covers ~30 degrees circumferentially and extends over the entire length of the active core region (4.26 m (14 ft)). The pads are contoured to minimize the impact on the downcomer annulus flow area and to reduce the probability of vortex generation in the downcomer.

Based on its review the staff determined that additional information was needed for the staff to complete its review. In RAI-SRP3.9.2-EMB1-02 the staff requested that the applicant discuss the potential fluid forces created by the redesigned neutron panels and their potential effects on the flow-induced vibration (FIV) excitation of the core barrel/core shroud. In its June 20, 2008, response, the applicant stated “The circumferential extent of the neutron panels was limited to correspond to the high vessel fluence levels, and thus minimize the flow blockage in the downcomer. The neutron panels are tapered circumferentially (following the reduction in fluence level) to minimize the flow area reduction. In addition, the reactor vessel inside diameter was increased by two inches over the core elevations when the panels were added. This results in a net flow area increase of 4 percent relative to the vessel-core barrel downcomer flow area before the panels were added. The lower average downcomer velocity is expected to offset the effects of the turbulence added by the neutron panels."

Based on its review, the staff finds that the applicant has provided a satisfactory explanation of how the additional effects of turbulence due to the neutron panels are neutralized. Therefore, the concerns related to RAI-SRP3.9.2-02 are resolved and the addition of the neutron panels is likely to have no detrimental effects.

3.9.2.3.2.3 Relocation of Radial Support Keys and Tapered Peripheral on the LCSP

The four lower radial support keys for the core barrel are currently located 45 degrees from the cardinal axes. There is also a spherical radius on the outer diameter of the LCSP. Core inlet flow distribution and reactor vessel pressure drop results from CFD computer analysis showed that the core inlet flow distribution and the reactor vessel pressure drop were acceptable with a 6-degree slope on the outer diameter of the LCSP. Having the slope instead of the spherical radius on the outer diameter of the LCSP results in sufficient room for the radial support keys to be relocated to the cardinal axes, which is the preferred location. This relocation of the radial support keys eliminates the potential for interference with the core shroud attachment studs and nuts at the 45-, 135-, 225-, and 315-degree locations.

Based on its review the staff finds that relocation of the radial support keys and providing a tapered surface instead of a spherical one has no detrimental effects and is, therefore, acceptable.

3.9.2.3.2.4 Downcomer Excitations and Related Responses

The nozzle region of the reactor vessel has not been changed so that the entering flow turbulence excitations do not change. The addition of the neutron panels and the increase in the inside (and outside) diameter of the reactor vessel over the core elevations, since the original calculations have been made, change the overall area of the downcomer slightly. The reactor vessel inside diameter below the nozzle has been increased. The flow area including the addition of the neutron panels, increased vessel diameter, and different specimen basket design is increased by approximately 4 percent. This tends to offset the turbulence and increase in local velocities generated by the presence of the neutron panels. Due to the addition of a flow skirt to the lower head of the reactor vessel, the excitations of the structures in the lower vessel head plenum are likely to be lower which also contribute to a lower core barrel vibration level.

Based on its review, the staff determined that additional information was needed for the staff to complete its review. Therefore, in RAI-SRP3.9.2-EMB1-10, the staff requested that the applicant provide analytical or test data to quantitatively validate this statement that the increase in the increase flow area by 4 percent is expected to offset the turbulence and increase in the local velocities generated by the presence of the neutron panels.

In its June 20, 2008, response, the applicant stated that all previous test data show that, for a given geometry and inlet flow pattern, the turbulence excitation decreases-usually by an exponent greater than 2-with decreased flow rate. The staff finds this response satisfactory and acceptable because the applicant has provided quantitative data to satisfy staff's concern. Therefore, concerns related to RAI-SRP3.9.2-EMB1-10 are considered resolved.

Based on its review, the staff finds that the changes in the vessel diameter, addition of the flow skirt and the presence of the neutron panels will have no detrimental effects on the downcomer excitations and related responses. These changes are, therefore, acceptable.

3.9.2.3.2.5 Reduction of Core Shroud Brace Thickness

Design modifications have been evaluated for the AP1000 core shroud subsequent to the analyses discussed above. The modification is to thin the core shroud braces to reduce thermal stresses. The staff concluded that this modification will not have a detrimental effect on the structural integrity of the core shroud and is therefore acceptable.

3.9.2.3.2.6 Reactor Coolant Pump-Induced Loads

RCP-induced forces are included in the responses reported in Section 7.7.2 of WCAP-15949-P Revision 2. A calculation to predict the pressure differences across the various reactor vessel internals components due to RCP pulsations was performed. However, since the original acoustic calculation using the ACSTIC code was completed, several design changes were made to the AP1000 reactor vessel and reactor vessel internals as discussed above. Specifically, the reactor vessel diameter was increased, the lower core restraints were relocated, neutron panels were added, specimen baskets were redesigned and relocated, and a flow skirt was added. To evaluate the impact on predicted pressure differences due to the previously noted design changes, an updated ACSTIC calculation was completed.

The updated calculation performs a similar analysis at hot full-power as the original calculation while considering the previously noted design changes. Additionally, the updated calculation also considers the hot functional test (HFT) conditions, including the absence of the core with 25 percent of the core pressure drop simulated near the exit of the LCSP. Consistent with the original calculations, three frequency ranges were evaluated with all RCPs in-phase and with two RCPs out of phase with the other two. The three frequency ranges are ±10 percent of the rotating speed frequency, the first blade passing frequency and the second blade passing frequency. The impact of the results of the updated calculation have been addressed in the individual component analyses for the guide tube, upper support column, core barrel, and core shroud.

The reactor internals were evaluated for the RCP startup conditions shown in Table 5-9a of WCAP-15949-P. The updated reactor conditions are shown in Table 5-9b of WCAP-15949-P. The updated conditions are less severe since the time to reach hot standby is the same for the new and old conditions but the flow rates during heat-up are lower for the new conditions. Therefore, fluid velocities are lower for the updated startup conditions than for the evaluated startup conditions. Lower flow rates would result in lower flow turbulence loads. Since the calculated high-cycle fatigue factors of safety are greater than one, the staff concluded that the AP1000 internals are adequately designed.

Based on its review as discussed above, the staff determined that it needed additional information to complete its review. Therefore, the staff requested that the applicant provide this information in the areas of concerns.

In RAI-SRP3.9.2-EMB1-01, the staff requested that the applicant describe the design and modeling of the core barrel/upper core plate as they relate to FIV structural dynamic analysis. The staff also requested that the applicant discuss the uncertainty associated with the modeling of the support interface employed in the modal analysis of the support. In its June 20, 2008, response, the applicant stated that the upper core plate is modeled as a part of the upper internals in the system model. The gaps between the upper core plate (and core shroud) slots and the alignment plates mounted on the core barrel are also modeled. To ensure that the entire range of possible gaps between the upper core plate and the core barrel alignment plates is evaluated, time-history analyses were performed with various sets of gaps (upper core plate, top core shroud plate, and core barrel lower supports). Table 6-9 in WCAP-15949-P, Revision 2 (Reference 1), shows the gaps modeled and the resulting loads. The resulting highest load was used in the structural analysis.

The staff finds the applicant's response reasonable and acceptable. Also, AP1000 DCD Section 3.9.2.3 was revised. Therefore, concerns related to RAI-SRP3.9.2-EMB1-01 are resolved.

3.9.2.3.2.7 Evaluation of WCAP-15949-P Revision 2

The staff's review and acceptance of WCAP-15949-P, Revision 1 is documented in Section 3.9.2.3 of NUREG-1793. The additional information in WCAP-15949-P, Revision 2 includes information to justify that there will be no impact on the vibration evaluation of the reactor internals as a result of the changes in the standard design. The staff's review in this safety evaluation includes this additional information. A preoperational HFT is to be carried out on the first AP1000 reactor internals, classified as a prototype, per requirements of RG 1.20, Revision 2. The AP1000 reactor internal design is the latest product of evolutionary changes to three-loop plants, starting with H. B. Robinson as the first prototype and the most recent ones being Doel 3 and Doel 4 (3XL), as described in Section 1.2 of WCAP-15949-P, Revision 2. The significant design changes in the AP1000 reactor internals relative to the Doel 3 and Doel 4 designs are described in Section 3 of WCAP-15949-P, Revision 2. The plant and scale model tests associated with each prototype (including the upper internal test of Doel 3 and the lower internal test of Doel 4) are summarized in Section 4, which also demonstrates the consistency among the various Westinghouse plant and scale model tests. The sources of the flow-induced vibration, considered in Section 5, of WCAP-15949-P, Revision 2 are the following:

- Flow turbulence
- RCP related
- Turbulence excitation of system fundamental acoustic mode
- Vortex shedding

In Section 5 of this WCAP, forcing functions simulating the various excitations are developed through correlation with the 3XL and other plant and scale model test data and put on AP1000 system models and sub-models. The results, in terms of peak stresses, on the various AP1000 critical components are presented in Section 6 and summarized in Table 2-1. The applicant has developed detailed CFD and finite-element models of both the 3XL and the AP1000 reactor vessel and internals designs as discussed in Sections 5 and 6 of this report. The 3XL finite-element model is used to calculate vibratory-induced deflections, and the calculated values are compared to applicable plant test data taken during the Doel 4 HFT. The finite-element modeling techniques are refined to accurately predict the Doel 4 test results, and these modeling techniques are applied in the AP1000 model. The CFD model was used to determine the steady-state flow loads on the upper internals components. Section 7 presents the detailed plan for the preoperational HFT and Section 8 presents the pre- and post-hot functional inspection program.

There is no instrumentation between the upper end of the core shroud and the LCSP. In RAI-SRP3.9.2-EMB1-03, the staff requested that the applicant discuss the rationale for and the location of instrumentation to provide predicted stresses and also provide the value and location of the maximum stresses for the core barrel/core shroud assembly. In its June 20, 2008, response the applicant stated, "A detailed description of the internals model is provided in WCAP-15949, Revision 2. The instrumentation is designed to provide adequate information to describe the vibration time histories and modal content. In the case of the core barrel, the beam modes can be inferred from the core barrel flange strain gages. The fundamental shell modes of the core barrel cover the entire length, the approximate midpoint being at the top of the core shroud where three radially sensitive accelerometers are mounted."

The staff finds the rationale for the panel location of the instrumentation reasonable and acceptable. With regard to the locations of the maximum stresses and adequacy of the instrumentation, the applicant stated…"the motions are defined by an assembly model. Where needed, sub-models are made to accurately define local, maximum stresses. Detailed core shroud models and sub-models are used to define maximum vibratory stress levels in the core shroud. Similarly, for the core barrel, models are used to define stresses at key locations such as core barrel flange (dominantly beam mode-induced stresses), and shell mode stresses) and barrel shell LCSP stresses (includes vertical motion-induced stresses). The strain gages and other transducers are located such that they are not in an extremely high gradient area and so that, with the analytical models they can adequately define the vibration so that maximum stresses can be determined from the analytical models. The maximum stresses for the core barrel/core shroud are provided in Table 2-1 of WCAP-15949. The maximum core barrel stress is at the core barrel wall to core barrel flange interface. The maximum core shroud stress is at the corner of the panel."

Based on its review of the above response, the staff finds that the instrumentation supported by the structural model (which is supported by the calculated versus measured mode shapes and natural frequencies) is adequate to define the maximum stresses due to flow and RCP-induced vibration. Therefore, the concerns related to RAI-SRP3.9.2-03 are resolved.

In WCAP-15949, Table 5.3, "Comparison of calculated and measured 3XL responses," it is stated that the accelerations are considered to be influenced by accelerometer pressure sensitivity and that vertical vibration content in the core barrel strain gages is difficult to ascertain because of masking by other contributors. Therefore, in RAI-SRP3.9.2-EMB1-04, the staff requested that the applicant discuss: (a) how the vibration content affects the strain gage data; (b) how associated conversion factors from 3XL to AP1000 are affected; and (c) the uncertainties in the conversion factors.

In its response, the applicant stated, "The strain gages are used to measure mean and oscillatory reactor internal responses. For example, in the core barrel flange strain gages, the oscillatory content includes contributions from core barrel beam modes, the vertical modes of the core barrel, and the shell modes of the core barrel. Supported by the core barrel analytical model and data from other transducers, the contribution of the various modes can be determined. This information is used to support the determination of the maximum stress in the core barrel flange.

During the 3XL hot functional vibration testing, it was observed that the accelerometer data included an unexpected magnitude of response at a particular frequency that was postulated to be due to system pressure pulsations. The accelerometer pressure sensitivity was confirmed by the accelerometer vendor. It is considered that this was adequately recognized in the interpretation of the 3XL data. The 3XL test data are used only to benchmark the analytical methods used to predict AP1000 responses, primarily the CFD based prediction of core barrel vibration. There are no conversion factors used in developing the AP1000 responses, since all of the AP1000 predictions are from analytical models."

Based on its review of the above response, the staff finds that the applicant has provided a satisfactory response to the staff's concerns related to how the vibration content affects the strain gage data, associated conversion factors from 3XL to AP1000 are affected, and the uncertainties in the conversion factors. Therefore, the concerns related to RAI-SRP3.9.2-EMB1-04 are resolved.

The overall methodology for estimating the vibration forces and using these forces to predict the response of the reactor internals is outlined in Figure 5-1 of WCAP-15949. In RAI-SRP3.9.2-EMB1-05, the staff requested that the applicant describe the methodology for determining bias errors and uncertainties associated with data obtained from various sources for evaluating AP 1000 reactor internals responses.

In its response, the applicant stated, "The transducers are calibrated prior to use. From this calibration, the voltage conversions at the temperature that the data were acquired are applied. Any uncertainty in the factors that convert voltages to physical units will also be recognized. It is also noted that expected and measured responses were similar in past tests. In view of these factors, it is considered that bias errors and uncertainties are less than the minimum margin to allowable values-presently 0.2 for AP1000 (per WCAP-15949-P, Revision 2, Table 2-1)."

The staff finds the applicant's explanation for justifying the bias errors and uncertainties as being less than 0.2 to be reasonable and satisfactory. Therefore, concerns related to RAI-SRP3.9.2-EMB1-05 are resolved.

NUREG-1793 discusses the evaluation of WCAP-15949-P, Revision 1 in Section 3.9.2.3. In RAI-SRP3.9.2-EMB1-06, the staff requested that the applicant discuss and summarize the significant additional information/items provided in WCAP-15949-P, Revision 2, dated June 2007.

In its response the applicant stated that the most significant changes between Revision 1 and Revision 2 of WCAP-15949 are the addition of the neutron panels, the reactor vessel diameter increase in the core region, the revised specimen basket arrangement, and the addition of a flow skirt to the reactor vessel. The overall conclusion that the vibration amplitudes are sufficiently low for structural adequacy of the AP1000 reactor internals has not changed. The applicant also provided an itemized list of changes between WCAP-15949-P, Revision 1 and Revision 2, in the RAI response. The staff reviewed this itemized list of changes and concerns related to RAI-SRP3.9.2-EMB1-06 are resolved.

Past experience related to testing of reactor internals indicates that instrument failures do occur during testing. Thus, it is prudent to provide redundancy in the data acquisition process. Therefore, in RAI-SRP3.9.2-EMB1-08, the staff requested that the applicant discuss the redundancy in the instrumentation proposed for the AP1000 reactor internals preoperational test program.

In its response the applicant stated, “Some redundancy is included in the number, location, and types of transducers installed during the Hot Functional Test program. For example both accelerometers and strain gages are installed on the core barrel, which provides some redundancy in the event that an individual transducer would fail.” In previous prototype tests conducted by the applicant, the instrument failures were not of sufficient quantity to preclude drawing the needed conclusions.

The transducers are installed on the reactor internals and subjected to known static and dynamic inputs prior to the HFT. These calibration tests relate displacements to measured strains and accelerations and this data is used to interpret the mean flow loads and flow-induced vibration amplitudes. The operability of these transducers is also verified during these static and dynamic calibration tests. In addition, some redundancy is included in the interpretation of the results in that a narrow band response centered on a particular frequency can be associated with a particular mode and the damping of that mode. This enables the stress distribution associated with this mode to be used to completely describe the stresses related to this mode.

Based on its review of the applicant’s response as discussed above, the staff finds that there is adequate redundancy in the instrumentation and satisfactory calibration procedures are in place. Therefore, the concerns related to RAI-SRP3.9.2-EMB1-08 are resolved.

In RAI-SRP3.9.2-EMB1-09, the staff requested that the applicant provide the following topical reports, which relate to preoperational test programs for the Trojan 1 and Doel 4 plants that are referenced in the AP1000 DCD Revision 17: (1) WCAP-8766, and (2) WCAP-10846. Additionally, the applicant was requested to provide test data from the core shroud at the Yonggwang 4 plant, which is relevant to the evaluation of the AP1000 reactor internals.

In its June 20, 2008, response, the applicant provided the two WCAP reports and the Yonggwang core shroud test report for staff’s review at the applicant’s Rockville, Maryland office. The staff reviewed these documents. The results of the Doel 3 and Doel 4 reactor internals vibration test programs were used to perform the vibration assessment of the AP1000 reactor internals. The measured responses from Doel 3 and Doel 4 were adjusted to the higher AP1000 flow rate to support the determination of the expected upper internals and lower internals vibration levels respectively. The velocity through the core is approximately the same as that of Doel 4. Based on its review the staff was satisfied that the applicant had used an acceptable methodology to perform the vibration assessment of the AP1000 reactor internals. The results of the Trojan 1 tests confirmed that the lower internals vibrations are lower with neutron panels than with a circular thermal shield as reported in WCAP-8766.

The staff is satisfied with the results, and concerns related to RAI-SRP3.9.2-EMB1-09 are resolved.

An acoustic analysis of the primary coolant loop has been provided in Section 5.1.3.1 of WCAP-15949. The impact of the results of the updated calculations has been addressed in the individual component analyses for the guide tube, upper support column, core barrel, and core shroud. The reactor internals were evaluated for the RCP startup conditions shown in Table 5-9a. The updated reactor conditions are shown in Table 5-9b of WCAP 15949. It is noted that the updated conditions are less severe since the time to reach hot standby is the same for the new and old conditions but the flow rates during heat-up are lower for the new conditions. Therefore, fluid velocities are lower for the updated startup conditions than for the evaluated startup conditions. Lower flow rates would result in lower flow turbulence loads. The applicant therefore concludes that there would be no overall impact due to the design changes.

In order to evaluate the impact on predicted pressure differences due to the design changes, an updated acoustic analysis using the computer code ACSTIC, was performed. However, simplifying assumptions were made in the acoustic modeling. The staff contended that the conclusions are not necessarily valid unless adequate justification is provided that the uncertainties associated with the ACSTIC calculation have been taken into consideration. In RAI-SRP3.9.2-EMB1-11, the staff requested that the applicant discuss how the uncertainties associated with acoustic analysis were factored into the results of the updated calculations.

In its response, the applicant stated, “The uncertainties associated with the ACSTIC calculation were considered by employing a general design basis in which the RCP-related responses are taken to be coincident with natural frequency if the natural frequency is within ±10 percent of the RCP excitation frequency. The calculated maximum forces from this resonance condition were then utilized in the reactor internals component structural evaluation.”

The staff finds the applicant’s response reasonable and acceptable, and concerns related to RAI-SRP3.9.2-EMB1-11 are resolved.

Based in its review of WCAP-15949-P, Revision 2, and Revision 17 of AP1000 DCD, Section 3.9.2.3, the staff finds that there is no overall impact due to the design changes.

3.9.2.3.3 Conclusion

This report supplements NUREG-1793 for the AP1000 standard plant design. NUREG-1793 was issued by the NRC in September 2004 to document the staff’s technical review of the AP1000 design. With the closure of OI-SRP3.9.2-EMB1-07 documented in this report, the staff concludes that the applicant has provided sufficient information to satisfy 10 CFR Part 50 Appendix A, GDC 1 and GDC 4 with regard to the dynamic testing and analysis of SSCs.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

3.9.2.4.1 Introduction

In Revision 16 to the AP1000 DCD, the applicant proposed to address COL Information Item 3.9-2 pertaining to irradiation-assisted stress-corrosion cracking (IASCC) and void swelling susceptibility evaluations for reactor internal core support structure materials.

In Section 3.9.2.4 of NUREG-1793, the NRC identified COL Action Item 3.9.2.4-1, in which the COL applicant will provide the design reports for the reactor internal core support structures including a final stress analysis conforming to the design provisions of the ASME Code, Section III, Subsection NG. The following section addresses the adequacy of the analyses for the reactor internals for IASCC and void swelling phenomena.

AP1000 Standard COL TR-12, APP-GW-GLR-035, Revision 0, was provided by the applicant under WCAP-16620-P, Revision 0, “Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant,” (hereafter designated as TR-12) dated July 31, 2006. TR-12 addresses AP1000 COL Information Item 3.9-2 pertaining to IASCC and void swelling in reactor internal core support structure materials for the AP1000 plant. COL

Information Item 3.9-2 corresponds to AP1000 DCD, Tier 2, Section 3.9.8.2 (DCD Section 3.9.8.2), Revision 15 and Action Item 3.9.2.4-1 from NUREG-1793. COL Information Item 3.9-2 is addressed in a proposed revision to DCD Sections 3.9.8.2 and 3.9.9. The staff reviewed the information provided in TR-12, including the proposed changes to DCD Sections 3.9.8.2 and 3.9.9. The revised DCD subsections are included in Revision 16 to the AP1000 DCD. The staff's findings regarding TR-12 are summarized below.

In TR-12, the applicant addressed the provisions of COL Information Item 3.9-2 pertaining to IASCC and void swelling susceptibility evaluations for reactor internal core support structure materials for the AP1000 plant. The applicant proposed to revise COL Information Item 3.9-2, in part, through the implementation of Revision 16 to DCD Section 3.9.8.2. In Revision 15 to the AP1000 DCD, Section 3.9.8.2, the COL Information Item stated:

> Combined License applicants referencing the AP1000 design will have available for NRC audit the design specifications and design reports prepared for ASME Section III components. COL applicants will address consistency of the core support materials relative to known issues of irradiation-assisted stress corrosion cracking and void swelling. [*The design report for the ASME Class 1, 2, and 3 piping will include the reconciliation of the as-built piping as outlined in subsection 3.9.3. This reconciliation includes verification of the thermal cycling and stratification loadings considered in the stress analysis discussed in subsection 3.9.3.1.2.*]

It should be noted that TR-12 only addresses the second sentence of DCD, Revision 15, Section 3.9.8.2. The other sentences in this revision to DCD Section 3.9.8.2 are addressed in separate AP1000 Standard COL TRs.

In Revision 16 to the AP1000 DCD, the applicant proposed to address the COL Information Item on a generic basis and revise Section 3.9.8.2 as it relates to IASCC and void swelling to state:

> The consistency of the reactor internal core support materials relative to known issues of irradiation-assisted stress corrosion cracking and void swelling has been evaluated and addressed in APP-GW-GLR-035 (Reference 21).

Revision 16 to DCD Section 3.9.8.2 specifically references TR-12 (i.e., APP-GW-GLR-035) as the technical basis for the evaluation of IASCC and void swelling phenomena in AP1000 reactor internal components. In addition to the above, Revision 16 to the AP1000 DCD adds the following reference (Reference No. 21) for TR-12 to DCD Section 3.9.9, "References":

> 21 APP-GW-GLR-035, "Consistency of Reactor Vessel Internal Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking and Void Swelling for the AP1000 Plant," July 2006.

3.9.2.4.2 Background

IASCC is an age-related degradation mechanism where materials exposed to high levels of neutron radiation become more susceptible to SCC with increasing neutron fluence. The current consensus is that susceptibility to IASCC is a significant concern for austenitic stainless steel and nickel-based alloy reactor internal components in both boiling-water reactors (BWRs) and PWRs. This is due to the fact that these components are exposed to elevated neutron fluence levels over the lifetime of the plant. The exact mechanisms for IASCC damage in reactor internal components are not well known. However, numerous studies suggest that IASCC results from the synergistic effects of irradiation damage to the material, changes in the local coolant-water chemistry, and the stress state in the component.

Irradiation-induced void swelling is an environmental degradation phenomenon that can affect reactor internal structural alloys exposed to high levels of neutron radiation. Void swelling is characterized by an increase in a component's volume due to the formation of voids as a result of neutron irradiation at elevated temperatures. Void formation occurs due to the migration and condensation of lattice vacancies in response to radiation-induced displacement of atoms from their lattice sites. Void swelling becomes more pronounced at higher structural temperatures due to higher diffusion rates. Some amount of swelling can occur in virtually all structural alloys under sufficiently high conditions of neutron fluence and temperature. However, austenitic stainless steels and nickel-based alloys, the primary alloys used in reactor internal core support components, are known to be susceptible to void swelling earlier and faster due to the multiple slip systems and close-packed nature of their face-centered cubic crystal structure. As many PWRs age, void swelling behavior in austenitic stainless steel and nickel-based alloy reactor internal components has become the subject of increasing attention. Excessive void swelling can lead to dimensional instability of the component and significant decreases in fracture toughness. It could also influence or contribute to the susceptibility of the component to IASCC, stress relaxation, and irradiation embrittlement.

3.9.2.4.3 EPRI Topical Report MRP-175

The U.S. Nuclear Power Industry is conducting ongoing studies of IASCC and void swelling phenomena in reactor internal structural components. The IASCC and void swelling data that have been accumulated thus far were summarized in a report issued by the EPRI Topical Report MRP-175, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)," dated December 2005. This report provided screening criteria and their technical bases for the age-related degradation evaluation of PWR reactor internal component items.

Appendix B of MRP-175 addressed IASCC in PWR reactor internal components and the establishment of an IASCC threshold and screening criteria for determining susceptibility to IASCC behavior. The report provided a comprehensive review of the open literature and industry operating experience regarding IASCC in American Iron and Steel Institute (AISI) Type 304 and 316 austenitic stainless steels; the differences in IASCC behavior of cold-worked versus solution-annealed SSs; and IASCC behavior in nickel-based alloys. In general, this review confirmed that IASCC may be a significant concern for reactor internal components during later stages in plant operating life. Although the exact mechanisms for IASCC are not yet known, the MRP-175 review cited numerous studies conclusively demonstrating that both the stress state in reactor internal components and radiation damage caused by increasing neutron fluence levels during plant service will result in increased susceptibility to IASCC. The review pointed to various studies indicating that radiation hardening is directly linked to IASCC. Radiation-induced segregation, a phenomenon of accelerated solute diffusion brought about by radiation-induced increases in vacancy concentration, was also cited as a possible contributor to IASCC. The IASCC studies and limited industry operating experience reviewed by MRP-175 were used as a basis for recommending IASCC screening criteria based on stress levels in the component and accumulated radiation-induced displacement damage, quantified in units of displacements per atom (dpa). For a given material exposed to specific radiation energy spectra, increasing neutron fluence values correlate directly with increasing dpa levels.

The MRP-175 review cited studies suggesting that thermo-mechanical history and chemical composition can potentially have a significant impact on IASCC resistance in austenitic stainless steel materials. In particular, cold-working has been shown to be potentially favorable for delaying the onset of radiation damage at lower damage levels (less than 10 dpa). This phenomenon has been attributed to the presence of a high density of dislocations for trapping radiation-induced point defects, thereby delaying the development of the microstructure responsible for radiation hardening. However, at higher damage levels (greater than 10 to 20 dpa), studies indicate that both solution-annealed and cold-worked materials attain the same degree of radiation hardening. Studies also indicate that differences in bulk alloy composition among various austenitic stainless steel reactor internal components can potentially have varying effects on IASCC initiation and progression. The higher nickel content of Type 316 was cited as a contributor to its greater resistance to radiation damage, compared with Type 304 stainless steel.

Oversize solutes such as titanium and niobium may also contribute to IASCC resistance by serving as trapping sites for point defects. Overall, MRP-175 concluded that, while IASCC susceptibility among various austenitic stainless steel materials is recognized to be affected by thermo-mechanical history and chemical composition, no consistent or quantitative correlation has yet been established. Thus, it was determined that a conservative set of IASCC screening criteria should be applied to all stainless steel alloys.

Section B.3 of MRP-175 stated that, based on numerous studies of IASCC phenomena, certain neutron fluence levels are a necessary precondition for the occurrence of IASCC in reactor internal components. For austenitic SSs, the MRP-175 review of data in the literature points to a conservative fluence threshold for IASCC in PWR reactor internal components of approximately 7×10^{20} n/cm^2 (E > 1.0 MeV), or a radiation damage level of about 1 dpa. However, the only known PWR IASCC incidents, observed in European PWR baffle bolts, have indicated an IASCC threshold level of approximately 2×10^{21} n/cm^2 (E > 1.0 MeV), or about 3 dpa. Additional evidence for the higher IASCC damage threshold was provided by studies which determined that IASCC initiation at 1 dpa can only occur under extremely high strain conditions (40 percent decrease in laboratory specimen cross section); such high strains are not representative of conditions in PWR reactor internal components. Further studies demonstrated that an IASCC damage threshold of 3 dpa existed for various heats of cold-worked 316 stainless steel, where stress levels in lab specimens exceeded the yield strength for the material. Based on these studies and the incidents that were observed in European PWR baffle bolts, the MRP-175 report concluded that 3 dpa represented a reasonable consensus estimate of the IASCC damage threshold for austenitic stainless steel reactor internal components. However, the MRP-175 report emphasized that, at the current time, the understanding of IASCC is not sufficiently advanced to suggest a definitive IASCC fluence or radiation damage threshold that is universally applicable to all PWR reactor internal materials.

Despite significant uncertainty regarding a precise IASCC threshold and the definitive prediction of IASCC susceptibility in PWR reactor internal components, the studies reviewed in the MRP-175 report point to a definite correlation of IASCC behavior with neutron fluence and stress levels in the component. Figure B-1 of MRP-175 presented curves, based on IASCC laboratory studies, depicting the stress level required for specimen failure by IASCC as a function of radiation damage, in dpa. A recommended IASCC screening curve was presented in Figure B-3 of MRP-175. This screening curve was derived by shifting the empirical curve for long term IASCC failure downward (to more conservative stress levels) to account for the observed baffle bolt failures in Europe. MRP-175 recommended that this lower bound IASCC screening curve be utilized at this time for developing IASCC screening criteria for PWR reactor internal components where radiation damage levels exceed 3 dpa.

Appendix G of MRP-175 addressed void swelling in PWR reactor internal components and recommended void swelling screening criteria. In general, MRP-175 found that void swelling may be a significant concern for reactor internal components in PWRs because it produces volume and dimensional changes that could potentially result in distortions within structural components as well as changes in fracture toughness properties. The MRP study of void swelling phenomena found that when volume changes in the material exceed approximately 5 percent, significant increases in embrittlement associated with the void swelling start to occur.

Furthermore, the MRP review of fast reactor data found that when volume changes in the material due to void swelling exceed 10 percent, the tearing modulus for 300-series stainless steels is dramatically reduced and falls to zero at room temperature, corresponding to severe embrittlement with little energy required for crack propagation.

Based on a comprehensive review of the literature and industry operating experience regarding void swelling behavior in austenitic stainless steels, MRP-175 concluded that void swelling behavior in reactor internal components is primarily influenced by structural temperature in the component and accumulated radiation damage (dpa level), with components becoming more susceptible to void swelling at higher temperature and damage levels. Studies also demonstrate that neutron flux (corresponding to the dpa rate) can affect void swelling behavior, with lower dpa rates resulting in greater swelling for a given accumulated dpa level. However, the effect of dpa rate on void swelling in PWRs has not been well quantified, and MRP-175 cited several other void swelling studies that did not observe a strong effect.

Numerous studies cited by MRP-175 have reported that other factors are known to affect void swelling behavior in reactor internal components. Void swelling data demonstrate that cold work has the beneficial effect of prolonging the void swelling incubation period, due to the elevated concentration of dislocations acting as traps for point defects in cold-worked materials. Chemical composition of stainless steel alloys is also known to affect void swelling behavior. For instance, nickel and chromium content strongly affect vacancy diffusivity, and therefore, the onset of void swelling. On this basis alone, Type 304 stainless steel always swells more than Type 316 with the same thermo-mechanical starting state. Stress is generally regarded as a factor that accelerates swelling, although it is not thought to be an important factor for most PWR applications. MRP-175 also pointed to various studies showing that a high helium content or helium production rate can affect void swelling behavior. Several studies suggest that the presence of preexisting helium gas bubbles may prolong the incubation period of void swelling under high dpa rates in fast reactors. This is thought to be due to helium gas bubbles acting as sinks for point defects, thereby delaying the onset of rapid swelling. However, under normal neutron irradiation conditions in PWRs, various studies have given conflicting results regarding the overall impact of helium on void swelling behavior in reactor internal components. For instance, helium atoms generated as a result of the transmutation of boron during irradiation can increase the swelling rate, as helium atoms combine with vacancy clusters, thereby facilitating void nucleation and growth. Furthermore, the production of helium gas bubbles in components during transmutation could have the net effect of increasing the overall swelling, thereby negating any beneficial effects of vacancy elimination.

MRP-175 suggested that screening of austenitic stainless steel reactor internal components for void swelling should be determined primarily by the structural temperature of the material, the accumulated dpa level, and the dpa rate that the material will experience during service. MRP-175 emphasized that the screening criteria should focus on the volume changes that occur as a result of void swelling behavior because embrittlement and distortion of the component, the primary structural consequence of significant void swelling, occurs as a result of these volume changes. MRP-175 cited numerous studies suggesting that the onset of void swelling-induced embrittlement occurs at a local void swelling percentage of approximately 5 percent. It was therefore recommended that void swelling of one-half this level (~2.5 percent) should necessitate further examination of the component. If it can be ascertained that local swelling in a component would never approach 2.5 percent, then void swelling is not a concern.

To date there have been no reports of PWR reactor internal components showing significant distortion or failures as a result of void swelling. The only PWR void swelling data comes from baffle bolts removed for IASCC evaluations. Very minor void concentrations were observed with transmission electron microscopy (TEM) in several baffle bolts removed from Point Beach, Unit 1; Farley, Unit 1; and Tihange (Belgium), Unit 1. MRP-175 summarized the results of these evaluations. The highest localized void fraction was estimated at 0.24 percent in one of the bolts removed from the Tihange plant. All other local void swelling measurements were significantly less, with half of the measurements showing no voids present. Furthermore, 0.24 percent void swelling would not be expected to significantly impact structural performance. Based on these data, MRP-175 determined that for austenitic stainless steel reactor internal components, localized regions with structural temperatures less than 320 °C (608 °F) and projected damage levels less than 20 dpa (~ 1.3×10^{22} n/cm^2, E > 1.0 MeV) would be expected to experience local void swelling levels of less than 2.5 percent. This was recommended as the preliminary criterion by which void swelling in the component may be ruled out. MRP-175 stated that localized regions in reactor internal components with structural temperatures greater than 320 °C (608 °F) and projected damage levels greater than 20 dpa (~ 1.3×10^{22} n/cm^2, E > 1.0 MeV) should be analyzed to determine the percentage increase in void fraction using the best currently available predictive equation developed by industry studies of void swelling behavior for 304 series stainless steel – Equation G-2 from MRP-175. This equation correlates the percentage increase in void concentration with temperature, dpa level, and dpa rate. If this equation yields a predicted void swelling percentage greater than 2.5 percent, then further functionality evaluations for the component are necessary.

3.9.2.4.4 Evaluation

The evaluation of AP1000 reactor internal components for potential susceptibility to IASCC and void swelling was addressed in TR-12. Section 1.2 of TR-12 provided a brief discussion of known issues of IASCC and void swelling in the currently-operating PWR fleet. The applicant indicated that reactor internal components in currently-operating Westinghouse plants have not exhibited significant IASCC or void swelling issues to date based on inservice inspections (ISIs) performed in accordance with the requirements of the ASME Code, Section XI. However, other PWR vendors have reported limited IASCC in reactor internal bolting applications for several PWR plants in Europe. Results from detailed inspections of cold-worked Type 316 stainless steel baffle bolts from Farley, Unit 1 (a the applicant three-loop design) showed no signs of cracking after 17 effective full power years (EFPY) of facility operation. The estimated neutron fluence exposure for these baffle bolts is 20 dpa.

Based on the IASCC studies and data that have been accumulated thus far, the known parameters directly affecting the onset and progression of IASCC in reactor internal structural components are peak stress level in the component and cumulative exposure to neutron radiation (neutron fluence) during plant service. For void swelling, the known parameters affecting its onset and progression are peak structural temperature in the component and neutron fluence. Therefore, screening of reactor internal components for potential susceptibility to IASCC and void swelling requires that these parameters be determined. Section 2 of TR-12 briefly discussed the calculation of these parameters for use in IASCC and void swelling screening evaluations. The applicant determined that IASCC screening would be based upon the peak stress to which a reactor internal component is subjected at full hot power. The peak stresses were said to be comprised of the “membrane stress intensity with additions due to bending and stress concentrations, steady state thermal stress additions, and high-cycle fatigue components.” The applicant stated that transients do not need to be considered for the IASCC stress calculations. The peak stress levels for each of the reactor internal components were provided in Table 2-1 of TR-12. The projected end-of-life (EOL) radiation damage levels for each of the reactor internal components were provided in Table 2-2. These damage levels were expressed in units of dpa. Table 2-3 listed the estimated structural temperatures for each of the reactor internal components during normal operation.

Section 3 of TR-12 discussed the screening of reactor internal core support structure components for potential susceptibility to IASCC. The components were evaluated through the use of a set of PWR-specific screening criteria based on stress state in the component and damage level. These screening criteria are essentially a set of threshold levels of damage level and stress, such that if the specific EOL damage level and structural stress levels for a given component are found to be below the screening criteria threshold levels, it could be concluded that IASCC would not be an applicable degradation mechanism for the component during the design life of the plant. Conversely, if the EOL damage level and structural stress levels for a component are found to be greater than or equal to the screening criteria threshold levels, IASCC is considered to be a potential degradation mechanism during the service life of the component. According to TR-12, satisfaction of the IASCC screening criteria (i.e., exceeding the stress and damage level threshold values) does not imply that IASCC will absolutely occur; rather it should be considered as a potential degradation mechanism.

The IASCC screening criteria used in TR-12 are as follows:

- For EOL damage level < 3 dpa, IASCC is not considered applicable for any stress conditions.

- For EOL damage level ≥ 3 dpa, IASCC may be applicable for specific ranges of damage level and stress. These ranges are defined as follows:

- For 3 dpa ≤ EOL damage level ≤ 10 dpa, IASCC is considered applicable if stress ≥ 427.5 MPa (62 ksi).

- For 10 dpa < EOL damage level ≤ 20 dpa, IASCC is considered applicable if stress ≥ 317.2 MPa (46 ksi).

- For 20 dpa < EOL damage level ≤ 40 dpa, IASCC is considered applicable if stress ≥ 206.8 MPa (30 ksi).

- For the three dpa ranges above, it is implied that if the component does not meet the applicable stress threshold, IASCC would not be considered applicable.

Table 3-1 of TR-12 evaluated the peak stress and EOL damage level for each of the reactor internal core support structure components against the above IASCC screening criteria to determine whether or not any of the components would be susceptible to IASCC. Although a number of components have a projected EOL damage level greater than 3 dpa, none of these components have peak stresses that exceed the IASCC threshold levels for stress listed above. It was therefore concluded that IASCC is not a potential degradation concern for the reactor internal core support structure components for the design life of the AP1000 plant.

Section 4 of TR-12 discussed the screening of reactor internal core support structure components for potential susceptibility to radiation-induced void swelling. The potential susceptibility of components was evaluated through the use of a PWR-specific screening criterion based on the structural temperature in the component during normal operation and EOL damage level. The void swelling screening criterion used in Section 4 of TR-12 is as follows:

> If the structural temperature for a component is greater than or equal to 320 °C (608 °F) during normal reactor operation, and the EOL damage level equals or exceeds 20 dpa, then void swelling has a potential to occur.

Section 4 of TR-12 invoked the criterion above to screen all reactor internal core support structure components for susceptibility to void swelling. Although several of the reactor internal core support structure components are listed as having either a structural temperature or an EOL damage level that is greater than the applicable threshold, none of the components were listed as having both structural temperature and EOL damage level greater than or equal to the above thresholds. Accordingly, the results of this screening led the applicant to the conclusion that none of the reactor internal core support structure components for the AP1000 plant are susceptible to void swelling for the design life of the plant.

Based on its initial review of the above information regarding the screening of AP1000 reactor internal components for potential susceptibility to IASCC and void swelling, the staff determined that additional information was required to complete its evaluation. In an RAI issued on January 18, 2007, the staff requested supplemental information concerning the IASCC and void swelling screening methodology. RAI questions 1, 3, 4, 5, 6, 8, 10, 11, 12, 13, and 14 addressed the IASCC screening methodology. RAI questions 2, 7, 9, and 15 addressed the void swelling screening methodology. The applicant provided responses to these RAI questions by letter dated May 2, 2007.

In RAI Question 1, part a (RAI 1a), the staff requested that the applicant clarify whether the IASCC and void swelling screening criteria were meant to be specific for the AP1000 reactor design or were meant to be applied to PWR environments, regardless of PWR design. In its response to RAI 1a, the applicant stated that the IASCC and void swelling screening criteria are generic for all PWR environments and may be applied to reactor internal components regardless of design. The staff found that this response adequately resolved RAI 1a because the applicant clarified the applicability of the IASCC and void swelling screening criteria.

In RAI 1b, the staff requested that the applicant confirm whether the IASCC screening criteria from Section 3 of TR-12 were established using the lower bound IASCC screening curve developed by EPRI in Figure B-3 of the MRP-175 report. In its response to RAI 1b, the applicant confirmed that the IASCC screening criteria in TR-12 were established using the lower bound IASCC screening curve developed by EPRI in Figure B-3 of the MRP-175 report. The staff found that this response adequately resolved RAI 1b because the applicant provided the requested statement regarding the bases for the IASCC screening criteria in Section 3 of TR-12.

In RAI 1c, the staff requested that, if the IASCC screening criteria in Section 3 of TR-12 were established based on the lower bound IASCC screening curve from Figure B-3 of the MRP-175 report, the applicant provide justification, based on environmental and material similarity, regarding how these IASCC screening criteria are applicable to reactor internal components for the AP1000. In its response to RAI 1c, the applicant stated that the materials specified for the AP1000 reactor internal components are similar to those used in the currently-operating Westinghouse three-loop extended length design. Operating parameters are also similar. IASCC screening of AP1000 reactor internal components was based on the same criteria (the lower bound IASCC screening curve from Figure B-3 of MRP-175) as those used for IASCC evaluations of reactor internal components in these operating reactors. Furthermore, the MRP-175 IASCC screening curve was developed as a generic lower bound curve for austenitic stainless steel reactor internal components in PWR environments, and its application was not intended for any specific set of material conditions (e.g., amount of cold-work, solution annealing, trace element composition). With respect to environmental similarity, the MRP-175 screening curve is based on radiation damage and stress level for the component, and according to the current understanding of IASCC, these are the two known environmental parameters directly affecting the onset and progression of IASCC behavior. Therefore, the IASCC screening curve in Figure B-3 of the MRP-175 report is applicable to the AP1000 reactor internal components, based on environmental and material similarity. Accordingly, the staff found that RAI 1c is resolved.

In RAI 1d, the staff requested that the applicant indicate whether reactor internal components that do not meet or exceed the IASCC screening criteria in TR-12 (i.e., components that do not meet or exceed the threshold stress and damage levels for IASCC) would ever be considered susceptible to IASCC. In its response to RAI 1d, the applicant stated that ongoing license renewal and life extension activities at operating Westinghouse reactors will develop new data concerning aging effects and aging management in reactor internal components. It is possible that new data may necessitate the consideration of IASCC in reactor internal components currently not considered susceptible to IASCC. However, at the present time, the IASCC screening criteria in Section 3 of TR-12 are applied for the purpose of determining whether or not a given AP1000 reactor internal component is susceptible to IASCC behavior during the operating life of the plant. Since none of the AP1000 reactor internal components have peak stress and EOL damage levels that meet or exceed the IASCC threshold levels from Section 3 of TR-12, none of the components are currently considered susceptible to IASCC. The staff found that this response adequately resolved RAI 1d because the applicant clearly stated how it applied the screening criteria for determining susceptibility to IASCC.

In RAI 2, the staff requested that the applicant confirm whether the void swelling screening criterion from Section 4 of TR-12 was established based on the void swelling screening recommendation developed by EPRI in Section G.7 of the MRP-175 report. The staff further requested in RAI 2 that the applicant provide justification, based on environmental and material similarity, regarding how the void swelling screening criterion is applicable to reactor internal components for the AP1000. In its response to RAI 2, the applicant confirmed that the void swelling screening criterion from Section 4 of TR-12 is based on the void swelling screening recommendation of MRP-175. With respect to the applicability of the MRP-175 void swelling screening recommendation to AP1000 reactor internal components, the applicant stated that the materials specified for the AP1000 reactor internal components are similar to those used in the currently-operating Westinghouse three-loop extended length design. Operating parameters are also similar. Screening of AP1000 reactor internal components for void swelling was based on the same criterion (the void swelling screening recommendation from Section G.7 of MRP-175) as that used for void swelling evaluations of reactor internal components in these operating reactors. Furthermore, the MRP-175 void swelling screening recommendation was intended to be generic for austenitic stainless steel reactor internal components in PWR environments, and its application was not intended for any specific set of material conditions (e.g., amount of cold work, solution annealing, trace element composition). With respect to environmental similarity, the MRP-175 void swelling screening recommendation is based on neutron fluence and peak structural temperature for the component, and based on the current understanding of void swelling, these are the two known environmental parameters directly effecting the onset and progression of void swelling behavior. Therefore, the void swelling screening recommendation from Section G.7 of the MRP-175 report is applicable to the AP1000 reactor internal components, based on environmental and material similarity. Accordingly, the staff found that RAI 2 is resolved.

In RAI 3, the staff requested further detail regarding how the peak stresses for the various reactor internal components in Table 2-1 of TR-12 were determined. The staff also requested, in RAI 3, that the applicant elaborate on why stresses arising from thermal transients were not considered in the peak stress calculations. In its response to RAI 3, the applicant stated that these stresses represented peak stress levels for normal operation. Finite element techniques were used in the computation of these stresses, and stress concentration factors were applied as appropriate. The reported stresses were intended to be conservative for IASCC screening of reactor internal components. With respect to consideration of thermal transients, the applicant indicated that the screening criteria stress levels (based on the MRP-175 IASCC screening curve) were developed for comparison with normal operating peak stress levels, and normal operating peak stress levels do not include stresses due to transient conditions. However, these stress levels do account for steady-state thermal stresses arising from temperature gradients within the reactor internal components during normal operation. The applicant emphasized that temperature gradients in reactor internal components are a steady-state phenomenon caused by the surrounding RCS temperatures and internal heat generation within reactor internal components due to gamma heating; these factors are known to result in steady-state temperature gradients and thermal stresses within reactor internal components during normal operating conditions. The staff found that this response adequately resolved RAI 3 because the applicant adequately clarified its methods for computing the peak stresses for the reactor internal components. Furthermore, the applicant conclusively defined these stresses as peak operating stresses that do not account for transient conditions and provided adequate justification for why transients were not considered in their computation. Therefore, the staff found that RAI 3 is resolved.

In RAI 4, the staff requested that the applicant define EOL for the projected radiation damage levels in Table 2-2 of TR-12 in terms of the total EFPY of facility operation. In its response to RAI 4, the applicant stated that EOL for the AP1000 design is considered to be 55.8 EFPY of facility operation. Therefore, the damage levels in Table 2-2 of TR-12 are projected out to 55.8 EFPY of facility operation. The staff found that this response adequately resolved RAI 4.

In RAI 5, the staff requested that the applicant discuss how ISI will be conducted for the reactor internal components during the operating life of the AP1000 plant. In its response to RAI 5, the applicant stated that ISI of reactor internal components during plant operating life will be driven by applicable codes and standards, as required by NRC regulations. At present, a VT-3 visual examination of all accessible surfaces of reactor internal core support structure components is required by the ASME Code, Section XI. These examinations must be conducted once during each 10-year ISI interval. Such visual examinations are currently performed using remotely controlled submersibles, underwater crawlers and/or pole-mounted cameras. The staff found that this response adequately resolved RAI 5 because the applicant adequately specified how ISI will be conducted for reactor internal components during the operating life of the AP1000 plant.

In RAIs 6 and 7, the staff requested that the applicant discuss how the EOL damage levels and estimated structural temperatures from Tables 2-2 and 2-3 of TR-12 were determined for the reactor internal components. In its response to RAI 6, the applicant stated that a radiation model of the reactor vessel and internal components was created and two distinct axial power distributions were utilized to determine damage levels in dpa. The higher damage level from the two core power distributions was listed for each reactor internal component in Table 2-2. In its response to RAI 7, the applicant stated that detailed finite element thermal calculations were performed to determine the structural temperatures reported in Table 2-3. These calculations accounted for the effects of gamma heating using two core power distributions. The distribution resulting in the highest component temperature was utilized and temperatures at localized regions within the components were evaluated. The highest localized temperature for the component during normal reactor operation was listed in Table 2-3. As with the peak operating stresses listed in Table 2-1, the structural temperatures listed in Table 2-3 represent peak temperatures during normal operation because the void swelling temperature threshold in Section 4 of TR-12 (based on the screening recommendation of MRP-175) was developed for comparison with normal operating temperature levels in reactor internal components. The staff found that these responses adequately resolved RAIs 6 and 7 because the applicant adequately clarified its methods for computing the EOL damage levels and structural temperatures from Tables 2-2 and 2-3 of TR-12. Furthermore, the staff found that these stated methods were appropriate for calculating temperature and damage levels for use in screening reactor internal components for IASCC and void swelling.

In RAI 8, the staff requested that the applicant discuss whether there are any localized areas within any reactor internal component that could be exposed to damage levels that exceed the IASCC screening criteria from Section 3.1 of TR-12. In its response to RAI 8, the applicant stated that the EOL damage level calculations accounted for localized areas in the reactor internal components. As such, the damage levels reported in Table 2-2 of TR-12 represent that maximum projected damage level based on the highest localized exposure in each component. Therefore, the staff found that RAI 8 is resolved.

In RAI 9, the staff requested that the applicant further explain how it screened certain reactor internal components for susceptibility to void swelling. Specifically, the staff noted that Section 4 of TR-12 concludes that void swelling is not a significant degradation mechanism for any of the reactor internal components in the AP1000 plant. This conclusion was apparently based on the fact that none of the reactor internal components met the void swelling screening criterion, as invoked in Section 4 of TR-12, which stated that if the structural temperature for a component is greater than or equal to 320 °C (608 °F) during normal reactor operation, and the EOL damage level equals or exceeds 20 dpa, then void swelling has a potential to occur. The staff reviewed the damage level projections and structural temperature levels listed in Tables 2-2 and 2-3 and noted that, while none of the components are listed as having both damage level and temperature greater or equal than the above temperature and damage level threshold values, several components are listed as having either temperature or damage level greater than the applicable threshold. Therefore, the staff requested that the applicant explain how it was determined that void swelling was not an applicable degradation mechanism for these components.

In its response to RAI 9, the applicant stated that the TR-12 void swelling screening criterion was based on the recommendations in the MRP-175 report and, as such, it requires that both temperature and damage level be greater than or equal to the above threshold levels. The staff did not agree with this interpretation of the void swelling screening recommendation from the MRP-175 report and, therefore, found that this response did not adequately resolve RAI 9. By letter dated July 11, 2007, the staff issued a second RAI on this subject in order to address screening of reactor internal components for void swelling where either temperature or damage level meet or exceed the above threshold levels. In this RAI, the staff indicated that the recommended void swelling screening criterion from the MRP-175 report was misinterpreted by TR-12 when applied to reactor internal components that met or exceeded only one of the two thresholds (temperature or damage level). The staff stated the position that void swelling may be a potential concern for reactor internal components if either temperature or damage level exceeds its applicable threshold. This position is justified because of the hypothetical situation where one of these parameters is significantly greater than the threshold, and the other is only marginally less. For such a situation, it would be unacceptable to dismiss the possibility of void swelling in the component only because just one the two thresholds had been exceeded. Therefore, the staff requested that the applicant justify why the several components that were listed in TR-12 as having either temperature or damage level greater than the applicable threshold were not deemed susceptible to void swelling.

In its second response to RAI 9, dated August 21, 2007, the applicant provided an analysis for demonstrating that there are no significant void swelling concerns for the components listed in TR-12 as having either temperature or damage level greater than the applicable threshold level. The applicant demonstrated that none of the components in question meet the hypothetical situation proposed by the staff, where one of the parameters (temperature or damage level) is significantly greater than the threshold, and the other is only marginally less. For the components with structural temperatures exceeding the 320 °C (608 °F) void swelling threshold, all of the EOL damage levels for these components are far below the 20 dpa damage threshold for void swelling, and the calculated structural temperatures are only slightly greater than the 320 °C (608 °F) threshold. One component, the core barrel inner wall, has a projected EOL damage level that is slightly greater than the 20 dpa threshold; however the calculated structural temperature is significantly less than the 320 °C (608 °F) threshold. The applicant further demonstrated that these components are extremely unlikely to experience any significant void swelling during the operating life of the plant by applying equation G-2 from MRP-175 for calculating the predicted void swelling percentage. Application of this void swelling equation to the dpa and temperature values listed in Table 2-2 and 2-3 of TR-12 and the dpa rate based on 55.8 EFPY of facility operation yields void swelling percentages of less than 0.10 percent for all of these components. MRP-175 recommended further examinations of reactor internal components for void swelling behavior are necessary only if the predicted void swelling percentage based on this equation, approaches 2.5 percent. Therefore, the applicant adequately demonstrated that void swelling is not a significant concern for any of these reactor internal components (or any other AP1000 reactor internal component) based on the current void swelling data and predictive models. Accordingly, the staff found that RAI 9 is resolved.

In RAI 10, the staff requested that the applicant reconcile differences between the 3 dpa damage threshold for IASCC susceptibility established in TR-12 and IASCC neutron fluence thresholds established in other reports. Specifically, the staff noted the IASCC neutron fluence threshold from WCAP-14577, “License Renewal Evaluation: Aging Management for Reactor Internals,” is 1×10^{21} n/cm^2 (E > 0.1 MeV). Additionally, the Babock and Wilcox report, BAW-2248, “Demonstration of the Management of Aging Effects for the Reactor Vessel

Internals," stipulates a neutron fluence threshold of 1-2 x 10^{21} n/cm^2 (E > 0.1 MeV). The staff noted in RAI 10 that, according to MRP-175, the 3 dpa threshold from TR-12 is roughly equivalent to an accumulated neutron fluence value of 2 x 10^{21} n/cm^2 (E > 1.0 MeV) for austenitic stainless steel materials. In its response to RAI 10, the applicant stated that 3 dpa was the recommended IASCC damage threshold from the MRP-175 report, and the MRP-175 screening criteria represent a consensus opinion of the EPRI MRP expert panel. The recommended 3 dpa damage threshold superseded the previous two the applicant reports. The staff found this response adequately resolved RAI 10 because MRP-175 determined that 3 dpa represents a reasonably conservative consensus value for an IASCC damage threshold for reactor internal components in PWR environments.

In RAI 11, the staff requested the applicant to discuss whether the 3 dpa damage threshold for IASCC in TR-12 was determined taking into consideration the effect of thermo-mechanical history (e.g., prior cold work, annealing, etc.) in reactor internal components that are exposed to neutron fluence levels less than 6.7 x 10^{21} n/cm^2 (E > 1.0 MeV). This question related to a statement from Section B.1.1 of MRP-175 referencing studies indicating that thermo-mechanical history may affect the onset of IASCC in reactor internal components exposed to these neutron fluence levels. In its response to RAI 11, the applicant stated the IASCC screening criteria, as applied to the reactor internal components in TR-12, are based on the screening recommendations of MRP-175. The MRP-175 IASCC screening recommendations are generic for all austenitic stainless steel materials in PWR environments. As such, the IASCC screening criteria and 3 dpa damage threshold were not developed based on any specific state of cold work (or any other prior thermo-mechanical preconditioning) in the material. While the amount of prior cold work had been shown to potentially delay the onset of IASCC at fluence levels less than 6.7 x 10^{21} n/cm^2 (E > 1.0 MeV), the applicant stated the AP1000 reactor internal components were screened using the MRP-175 screening recommendations without regard to the components' thermo-mechanical history. The applicant further stated it is not anticipated that a material's degree of cold work will necessitate screening criteria be different from the criteria recommended by MRP-175. The staff found this response adequately resolved RAI 11 because the applicant adequately explained why the MRP-175 IASCC screening recommendations were applied irrespective of the thermo-mechanical history of the components. The staff's justification for acceptance of MRP-175 recommendations for generic screening of AP1000 reactor internal components for IASCC (irrespective of the components' thermo-mechanical history) is provided below.

In RAI 12, the staff requested that the applicant discuss whether the 3 dpa damage threshold for IASCC in TR-12 was determined taking into consideration the effect of differing chemical composition for the various reactor internal components. This question related to a statement from Section B.1 of MRP-175 referencing studies indicating differences in bulk alloy composition of elements such as silicon, nickel, niobium, titanium, and boron, among various austenitic stainless steel reactor internal components can have varying effects on IASCC initiation and progression. In its response to RAI 12, the applicant stated the IASCC screening criteria, based on the generic screening recommendations of MRP-175, did not consider variations in the elemental composition among the various reactor internal components. As with the case above concerning the potential effect of components' thermo-mechanical history, the staff found this response adequately resolved RAI 12 because the applicant adequately explained why the MRP-175 IASCC screening recommendations were applied irrespective of the components' specific elemental composition. The staff's justification for acceptance of MRP-175 recommendations for generic screening of AP1000 reactor internal components for IASCC (irrespective of the components' specific elemental composition) is provided below.

In RAI 13, the staff requested that the applicant discuss whether the 3 dpa damage threshold for IASCC in TR-12 is applicable to reactor internal components fabricated from nickel-based alloys, such as alloy X-750 and Alloy 690. In its response to RAI 13, the applicant indicated IASCC studies reviewed in MRP-175 have shown that the IASCC resistance of nickel-based alloy X-750 is approximately the same as for Type 304 and 316 austenitic SSs. Furthermore, AP1000 reactor internal reactor internal components fabricated using nickel-based alloys will be exposed to a projected EOL damage level of, at most, 0.04 dpa. Since this damage level is far below the 3 dpa IASCC damage threshold, IASCC is not considered to be a relevant degradation mechanism for these components. The staff found this response adequately resolved RAI 13 because the applicant adequately addressed IASCC screening of reactor internal components fabricated from nickel-based alloys.

Studies have shown crevice corrosion may be enhanced in reactor internal components due to the production of oxidizing ions in component crevices during exposure of reactor coolant to neutron radiation. Therefore, in RAI 14, the staff requested that the applicant discuss whether the effects of crevice corrosion were taken into consideration in screening AP1000 components for IASCC. In its response to RAI 14, the applicant stated the IASCC screening criteria do not explicitly address the effects of crevice corrosion in reactor internal components. However, crevice corrosion is prevented or controlled in AP1000 reactor internal components through the use of hydrogen overpressure, which minimizes the adverse effects of any oxygen that may be present due to heat-up or cool-down of the reactor system. Furthermore, crevice locations in AP1000 reactor internal components have been designed to allow flushing to prevent stagnation, a key contributor to crevice corrosion. The staff found this response adequately resolved RAI 14 because the applicant addressed how crevice corrosion would be mitigated in AP1000 reactor internal components.

Transmutation products such as helium are known to play an important role in void swelling. In order to reduce overall interfacial energy, helium atoms will combine with vacancy clusters, thereby facilitating void nucleation and growth. Section G.1 of MRP-175 states a potentially important aspect of void swelling in PWRs arises from transmutation of trace amounts of boron, preexisting in most austenitic SSs, to produce lithium and helium. Section G.1 of MRP-175 indicates at low neutron exposure (~10^{21} n/cm^2 thermal), almost all Boron-10 (20 percent of natural boron preexisting in trace quantities in most SSs) will be converted to lithium, producing helium in the process. Since the original concentration of boron in austenitic stainless steel reactor internal components is not generally reported in certified material test reports, it is difficult to assess the concentration of helium in the reactor internal components. In RAI 15, the staff requested that the applicant address whether the void swelling screening criterion from Section 4 of TR-12 accounts for the effects of helium on void swelling in stainless steel reactor internal components. In its response to RAI 15, the applicant stated the void swelling screening criterion, based on the generic screening recommendations of MRP-175, did not explicitly consider the effects of helium. The staff found this response adequately resolved RAI 15 because the applicant explained the MRP-175 void swelling screening recommendations were applied irrespective of the components' helium content. The staff explained its acceptance of the void swelling screening evaluation for the AP1000 reactor internal components (irrespective of the components' potential helium content) in the discussion of the applicant's responses to RAI 9 above.

The acceptance of MRP-175 screening recommendations would provide a basis for setting IASCC screening criteria in TR-12. There are currently limited data to support an all-encompassing set of IASCC screening criteria that can be generally applied to reactor internal components in PWRs. Furthermore, MRP-175 has referenced studies showing variability in chemical composition, microstructural characteristics, and thermal-mechanical history between similar alloys may result in differing stress and fluence thresholds for IASCC. MRP-175 cited numerous documents both in the nuclear power industry and the open literature identifying a variety of possible threshold values for IASCC susceptibility and, therefore, a definitive, all-encompassing set of IASCC screening criteria is not likely to exist. In its response to the staff comments regarding these issues, EPRI acknowledged that exact threshold values for IASCC are expected to depend on variables, such as chemical composition, microstructural properties, and thermo-mechanical history. However, EPRI stated the IASCC screening recommendations of MRP-175 represent a consensus based on the limited amount of available data, and the IASCC screening criteria are considered to be conservative for general application to IASCC evaluations of reactor internal components in PWRs. As such, MRP-175 concluded the IASCC screening criteria were appropriate for evaluating stainless steel reactor internal components to determine their susceptibility to IASCC behavior.

The staff found the limited amount of data does support the MRP-175 conclusions regarding the conservatism of the IASCC screening criteria from Section B.3 of MRP-175. Therefore, although it may be impossible to absolutely rule out the possibility of IASCC, because reactor internal components are deemed not susceptible according to the MRP-175 screening criteria, significant IASCC behavior would not be expected for the AP1000 reactor internal components because the peak operating stresses and projected EOL damage levels for these components fall significantly below the MRP-175 screening criteria threshold levels. Furthermore, any age-related degradation of reactor internal components due to IASCC would be gradual, and the ASME Code, Section XI requirements for ISI of reactor internal components will be sufficient for capturing any age-related degradation that may occur due to IASCC phenomena.

Based on the above considerations, the staff determined that the applicant had adequately addressed the staff's concerns, as documented in the above RAIs, regarding the IASCC and void swelling screening methodologies. Therefore, the staff found the applicant had appropriately evaluated the AP1000 reactor internal components for susceptibility to IASCC and void swelling in TR-12. Furthermore, the staff agreed with the conclusions in TR-12 regarding the determination that IASCC and void swelling are not projected to be significant degradation concerns for the reactor internal components in the AP1000 plant.

The staff determined the TR-12 conclusions regarding the evaluation of reactor internal components for IASCC and void swelling meet the requirements of ASME Section III based on the MRP-175 screening criterion as reported in TR-12 and is fully represented in Sections 3.9.8.2 and 3.9.9 of the AP1000 DCD, Revision 16. Therefore, the staff found the DCD changes, as proposed by the applicant in TR-12, acceptable and AP1000 COL Information Item 3.9-2 is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes incorporated into Revision 16 contribute to the increased standardization of the certification information in the AP1000 DCD and, thus, meet 10 CFR 52.63 (a)(1)(vii).

3.9.2.4.5 Conclusions

The staff finds the evaluation of the AP1000 reactor internal components for IASCC and void swelling meets the requirements of 10 CFR 50.55a by meeting the ASME Section III based on the MRP-175 screening criterion as reported in TR-12 and resolution of IASCC aspects of COL Information Item 3.9-2 is acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Introduction

The staff evaluation was first performed for AP1000 DCD, Revision 16, and TR-134, Revision 5, which were issued as part of AP1000 DC amendment application. The staff subsequently included AP1000 DCD Revision 19 in its evaluation when this revision was issued by the applicant.

3.9.3.2 Evaluation

DCD Tier 2, Section 3.9.8.2, addresses the combined license information for the design specifications and reports for the major ASME Code, Section III components. In DCD Tier 2, Revision 17, COL Information Item in Section 3.9.8.2, the applicant stated the design specifications and design reports for the major ASME Code, Section III components were available for NRC audit via the TRs listed in Table 3.9-19. It is also stated that design specifications and selected design analysis information were also available for ASME Code, Section III valves and auxiliary components. The applicant's letter dated February 8, 2008, states that design specifications and design reports for most major components and auxiliary equipment and valves would be available for NRC review in July 2008. In RAI-3.9.3-EMB2-01, the staff requested that the applicant verify the schedule provided in the letter was still valid.

In a letter dated June 26, 2008, the applicant stated review of design specifications and as-designed design reports for ASME Code Section III components provides a means for the NRC to verify the design commitments in the DCD are being implemented appropriately. This review permits some level of verification during the review of the COL applications. The ultimate check on the proper implementation of design requirements for ASME Code Section III components are the ITAAC require as-built design reports for the ASME Code, Section III components. The applicant stated that it has a substantial amount of design information available for NRC review. This information was sufficient for the NRC to start its review and support the conclusion that the ASME Code, Section III components were in compliance with the commitments in the DCD. The remaining design information needed to complete the NRC verification of the design criteria and methodologies in the DCD would be available in the short term consistent with the schedule for review of the AP1000 DC amendment.

The applicant stated since the information needed to address the COL Information Item on design specifications and design reports would be complete during the review of the AP1000 Design Certification amendment there was no need for a DAC or design ITAAC on the design of ASME Code Section III components. The applicant revised its approach of resolving the component design issue, and stated the revision of the COL information item in DCD Revision 16, Section 3.9.8.2, was based on the expectation the design information available at the time was sufficient for the NRC to reach a conclusion as to the implementation of the design requirements. The applicant stated that the amount of information available for NRC review was much more developed and robust. The COL information would be revised to reflect that sufficient information would be provided to the NRC to complete its verification of the implementation of the design commitments. The applicant stated it expected that this portion of the COL information item would be satisfied and no additional information would be required of the COL applicant. The portion of the COL information item, restated as a COL holder item, is related to the as-built reconciliation of thermal cycling and stratification loadings on piping.

The applicant stated it had completed the design specifications for the major ASME Code, Section III components in the AP1000. The as-designed design reports and supporting analysis for most of the major components were also complete and available for review. The applicant stated the analyses include the use of the updated (six soils case) seismic design spectra. The components that had or will have as-designed design reports ready for review on a schedule to support the NRC preparation of the SER with open items include: reactor vessel, control rod drive mechanism, steam generator, pressurizer, passive residual heat removal heat exchanger, core makeup tank, accumulator, and reactor internals. The applicant stated all of the design specifications for valves and auxiliary equipment are now available for NRC review or will be ready in time to support the preparation of the SER. This includes the motor-operated globe and gate valves. The balance of the design specifications (expected to be one or two) will be available for review well before open items need to be cleared for the advance SER.

The staff has reviewed the above additional information provided by the applicant, and found the commitment and schedule for resolving the COL Information item to be acceptable. Pending a successful audit for the required design specifications and design reports, the staff would be able to conclude whether the COL information item is closed.

In its letter of June 28, 2008, the applicant provided a markup of the revised DCD Section 3.9.8.2, which now states the following:

> The design specification and design reports for the major ASME Code, Section III components and piping made available for NRC review are identified in APP-GW-GL-002. Design specifications for ASME Code, Section III valves and auxiliary components made available for NRC review are also identified in APP-GW-GL-TBD

In doing so, the applicant has proposed to delete Table 3.9-19 from DCD, Tier 2, which lists the TRs summarizing design specification and design reports for ASME Code Section III components and piping. The applicant also deleted references to Reference Items 22 through 32 in DCD Section 3.9.9, and placed a new Reference 22, APP-GW-GL-002, "Design Specifications and Design Reports for ASME Code, Section III Components and Piping."

The staff has reviewed the above information provided by the applicant and concludes that the additional information provided by the applicant is acceptable in responding to the staff's request of RAI-3.9.3-EMB2-01. Subsequently, the staff reviewed AP1000 DCD, Revision 17, when it became available, and verified the latest revision of DCD has incorporated all the changes as required.

During October 13 to 17, 2008, the staff conducted an onsite review of the AP1000 component design in relation to the close out of the above COL Information Item in DCD Section 3.9.8.2. The purpose of the on-site review was to verify the AP1000 component design was in accordance with the methodology and design criteria described in the DCD, and satisfies the guidance provided in NUREG-0800 Section 3.9.3 for design specifications and design reports. This includes verification the design information described in the DCD was adequately translated into documentation for each of the components designed to ASME Code Section III, Class 1, 2, and 3 requirements. A separate staff audit report, dated August 3, 2009, documents the detailed on-site review for the design of the AP1000 mechanical components, including valves. During the audit, the staff identified concerns with the reactor vessel J-groove weld design and the additional details on the containment recirculation screen design.

The staff requested in RAI-SRP-3.9.3-EMB2-05 that the applicant demonstrate how the Westinghouse methodology meets the ASME Code for the J-groove weld design.

In response to RAI-SRP-3.9.3-EMB2-05, the applicant stated it had satisfied the intent of Paragraph NB-3228.5(a) of the ASME Code, Subsection NB. According to the applicant, the purpose of the Paragraph NB-3228.5 is to limit potential excessive distortion due to incremental plasticity, sometimes referred to as stress ratcheting. The location where this applies is the J-groove weld between the piping penetration and reactor vessel head. The overstress shown in the design report is caused by the large hoop stress combined with, to a lesser degree, the axial stress. The ratcheting mechanism cannot occur as a result of the hoop stress since it is restrained by the reactor vessel head. The stresses in the radial and axial directions are well within the limits and meet the ASME Code requirements. Therefore, according to the applicant, additional plastic analysis in accordance with Paragraph NB-3228.4 was not necessary.

The staff found that the applicant's response did not resolve the issue, and again asked the applicant to provide additional information or detailed information to demonstrate the J-groove weld design meets the ASME Code requirements. The staff's concern was that the design report for the reactor vessel head penetrations split the stress components at these locations to justify the satisfaction of the Code requirements. Paragraph NB-3228 is based on stress intensities and does not allow splitting stresses for the purpose of satisfying the ASME Code.

The applicant in its response stated that the justification previously provided for meeting the requirements of NB-3228.5 is compatible with ASME Code methodology. According to the applicant, the fatigue evaluation for stresses is made for a plane of reference. The fatigue evaluation checks the range of stress intensity values for every potential plane (line) of failure and fatigue usage is determined for a point on that plane using a conservative value of primary plus secondary stress intensity range, for the purpose of determining a conservative value of Ke and, therefore, a conservative usage factor. As this conservative approach does not satisfy the limits of Paragraph NB-3228.5(a), the Code rules are used to perform a more realistic evaluation using membrane and bending stresses normal to the plane of reference. Using this approach, Paragraph NB-3228.5(a) is met with very large margin. This approach is within the Code rules specified in the Code definitions. This evaluation therefore demonstrates compliance with Paragraph NB-3228.5 and a plastic analysis is not required. The reactor vessel design report and associated stress calculation for the vessel head penetrations will be revised with this discussion.

During April 19 to 21, 2010, the staff conducted an onsite review of Open Item OI-SRP-3.9.3-EMB2-05, "Component Supports, and Core Support Structures." The applicant, in its plastic analysis of control rod drive mechanism (CRDM) and vent pipe penetrations, has demonstrated that the design of the vessel head assembly satisfies the ASME Code requirements. On this basis, the staff finds this acceptable and Open Item OI-SRP-3.9.3-EMB2-05 is closed. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text. In addition, APP-MV01-Z0C-015, "Detailed Analysis of Closure Head and Vessel Flange Region for AP1000 Reactor Pressure Vessel," and APP-MV01-Z0C-019, "Detailed Analysis of Closure Head Penetrations (CRDM, UMI, and Vent Pipe) for AP1000 Reactor Vessel" have been updated to include the results of the plastic analysis for the CRDM and vent pipe penetrations.

The staff reviewed the design specification and other supporting documents associated with containment recirculation screens and found several issues are not completely addressed in the design specification. In RAI-SRP-3.9.3-EMB2-08, the staff requested that the applicant address the following:

(a) According to the design specification, the supplier will provide additional design details, design drawings and requirements. Therefore, the engineering drawings (envelope drawings) of the screen assemblies were not available at the time of site audit or at the Rockville office. Provide these engineering drawings of the screen assemblies for review by the staff.

(b) Provide the following loading conditions and combinations: (i) design and service level A-D loads and load combinations, (ii) fatigue evaluation, and (iii) the origin and the basis of using ±5 psi pressure loading on the IRWST screen from sparger discharge.

(c) Justify the latent debris mass value used for the screen pressure drop component of the structural load on the IRWST and sump screens. Additionally, verify the flow rate through the screen is conservatively calculated.

During April 19 to 21, 2010, the staff conducted an onsite review of Open Item OI-SRP-3.9.3-EMB2-08. As a result of the audit, the staff identified follow-up items that required the applicant's response:

1. Screen design reports and detailed design drawings were not available, since they have not yet been provided by the responsible vendor. A set of drawings was reviewed by the staff, but these are categorized as "Envelope Drawings" and not detailed design drawings. However, AP1000 DCD, Tier 1, Table 2.2.3-4, ITAAC 5.a) was added to require review of a report verifying the as-installed screens including seismic load, post accident operating loads, head loss and debris weights.

   Subsequently, the staff confirmed the existing AP1000 DCD, Tier 1, Table 2.2.3-4, ITAAC 5.a) addressed the issue such that this ITAAC allows the staff to inspect the design reports and verify the as-installed screens including seismic load, post accident operating loads, head loss and debris weights. The staff concurred with the ITAAC approach of screen design report. The staff finds that these audit findings satisfied the staff's request for the information in part (a) RAI-SRP-3.9.3-EMB2-08 and follow-up item #1. Therefore, the follow-up item number 1 is closed.

2. The staff questioned the loading on the screen (i.e., how the 0.25 psi pressure drop loading will be added to the 34.5 kPa (5 psi) loading for the screens). Following discussion, the applicant agreed the screen head loss component of pressure loading on both the IRWST and Containment screens would be a minimum of 1.72 kPa (0.25 psi) as indicated in a Westinghouse design specification (APP-GW-GLE-002, "Impacts to the AP1000 DCD to Address Generic Safety Issue GSI-191," February 2010, p. 39). In addition, the applicant stated that it would consider augmenting this minimum head loss to allow for additional margin.

   In response to this follow-up item, documented in the applicant's response to OI-SRP3.9.3-EMB2-08, Revision 1, the applicant stated it is very unlikely that sparger actuation will occur coincident with IRWST injection as the sparger actuation comes from the ADS 1, 2, 3 discharge. IRWST injection cannot occur until the pressure in the RCS drops below the hydrostatic pressure exerted on the IRWST water due to the level in the tank. This only occurs after ADS 4 has been actuated. Therefore the 1.72 kPa (0.25 psi) debris differential pressure load is not coincident with 34.5 kPa (5 psi) loading for the screens. These two loads are not included in the same load combination. The staff finds the response satisfies the staff's request for the information in part (b)(iii) RAI-SRP3.9.3-EMB2-08 and the follow-up item number 2. Therefore, the follow-up item number 2 is closed.

3. The staff asked the applicant to confirm the applicability of the 5 psi sparger loading on IRWST screen design. The operation of the IRWST tank spargers leads to a pressure loading on the IRWST screens. An estimate of the sparger pressure loading is used as one component of the loading of both the IRWST and the containment screens. During discussions, the staff asked the applicant for a review of the available documents that support the pressure loading of ±34.5 kPa (±5 psia). The applicant responded by saying the documentation for this was not available at the time. However, the applicant stated the documentation for the magnitude of the sparger pressure loading was available, and that staff had this in their possession.

   In response of this follow-up item, documented in OI-SRP3.9.3-EMB2-08, Revision 1, the applicant stated the "Sparger Loading" is the maximum value due to actuation of the sparger. The actual shape of the pressure will be sinusoidal shape. The forces in the walls of the IRWST are bounded by a case with a uniform pressure of 5 psi applied to the walls. The actuation of the sparger will occur during discharge of ADS 1, 2, 3 valves. Tests conducted at the ENEA's VAPORE facility showed the maximum pressure exerted on the IRWST walls during a sparger actuation of 181.r kg/s (400 lbm/s) steam. The pure steam blowdown caused the highest pressures exerted on the IRWST floor directly below the sparger arm during sparger actuation. Additionally, the tests simulated a sparger steam flow of 181kg/s (400 lbm/s). This flow more than bounds the actual calculated maximum steam flow of 65.8 kg/s (145 lbm/s) for the AP1000 (APP-ME02-Z0C-001 Revision 0). The nominal hydrodynamic load exerted during the above mentioned sparger test was 34.5 kPag (5 psig). Given the steam flow was more than 2.7 times the actual design flow, this bounds the structural design requirement for DP mentioned above. The staff finds the response satisfies the staff's request for the information in part (b)(iii) of RAI-SRP3.9.3-EMB2-08.

   The staff confirmed these support documents are available for NRC review and verified the pressure loading of ±34.5 kPa (±5 psia) was acceptable by NUREG-0800 Section 6.2 evaluation. The staff concurred with the applicant's response of the pressure loading of ±34.5 kPa (±5 psia). Therefore, this follow-up item is closed.

4. The staff questioned the potential sloshing of water in the IRWST tank resulting from seismic activity and the magnitude of resulting pressure loading on the IRWST screen structures. The applicant did not have a response to whether this potential source of loading had been considered. The follow-up item for the applicant is to determine if the sloshing load on the screen should be included.

   In response to this follow-up item, documented in OI-SRP3.9.3-EMB2-08 R1, the applicant stated the sequences for the postulated accidents are such that the seismic event producing the sloshing and the actuation of the sparger are not coincident. Therefore, the sloshing loads need to be included as a load on the screen, but it is not combined with the sparger loadings. The staff finds the response satisfies the staff's request for the information in part (b)(i) of RAI-SRP3.9.3-EMB2-08. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5. In response to part (c) RAI-SRP3.9.3-EMB2-08, documented in OI-SRP3.9.3-EMB2-08 R1, the applicant stated that the AP1000 debris inventory in containment and the development of supporting containment cleanliness programs have been reviewed by the NRC. The screen pressure drop and the flow rate through the screen have been developed in support of responding to GSI-191 issues. This information is documented in APP-GW-GLN-147, "Screen Design Report," and APP-GW-GLR-79, "Verification of Water Sources for Long Term Recirculation Following a LOCA."

The staff confirmed these support documents are available in NRC review, verified AP1000 debris inventory in containment and the development of supporting containment cleanliness programs were acceptable by NUREG-0800 Section 6.2 evaluation. The staff concurred with the applicant's response of part (c) RAI-SRP3.9.3-EMB2-08. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The AP1000 component design has been completed to the extent that the COL Information Item in Revision 15 is not necessary, allowing the aspects of the COL information item addressing components to be eliminated, as documented in the amendment to the DCD.

Since its original DC amendment, the applicant has modified the AP1000 seismic design ground motion requirements, in order to extend the DC application to soil sites. It was expected that these revised seismic loadings would have an impact on the component designs already performed up to that point. In RAI-SRP3.9.3-EMB2-02, the staff requested that the applicant confirm that, for all the major ASME Code Section III, components already designed, all the pertinent design specifications and design reports have been updated to incorporate the effects of the newly modified seismic loadings. In a letter dated June 28, 2008, the applicant changed the design basis for the major ASME Code, Section III components to include the design spectra and seismic requirements envelope for the HR and associated with the expanded soil conditions (six soils case). These revised seismic design requirements for the six soils case are included in the design specifications for the major ASME Code components. The applicant stated the analyses supporting the as-designed design reports prepared or being completed for NRC review were in compliance with the design specifications and include these revised seismic requirements of the six soils case. Based on the above response and the confirmation obtained from the staff's onsite review, the staff found the applicant has adequately incorporated the latest revised seismic input motions for the component design. RAI-SRP3.9.3-EMB2-02 is, therefore, closed.

In APP-GW-GLR-115, Revision 2, Section 6.2.2, Tables 6.2.2-2, 6.2.2-3, 6.2.2-5 provided the resultant forces of three sample components of the primary coolant loop. The analysis of these components were to evaluate the comparison of the loads from the HRHF input to those obtained from the time history associated with the CSDRS input. For the three sample components, the applicant indicated the resultant forces due to CSDRS excitation are higher than the resultant forces caused by the HRHF excitation. Therefore, the HRHF loads will not govern the component design, but the CSDRS loads will. The staff found the loads of HRHF excitation and seismic loads to components were adequately incorporated in Revision 2 of the APP-GW-GLR-115 report and are acceptable.

The staff's review of the applicant's evaluation on the effects of high frequency seismic input on the AP1000 mechanical component design is provided in Section 3.10 of this report.

3.9.3.3 Conclusions

Based on the information provided in the applicant's responses to the RAIs, the staff finds that the application meets the guidance of RG 1.206, NUREG-0800 Section 3.9.3 and the regulations for the design of mechanical components, and is acceptable. Piping-related issues are discussed in Section 3.12 of this report.

3.9.4 Control Rod Drive Systems

In Revision 17 to the AP1000 DCD, the applicant proposed changes to the hydrostatic test pressure for the CRDM housing as well as other materials related to changes to the CRDM. This resulted in changes to the DCD in Sections 3.5.1.2.1.1, 3.9.4.1.1, 3.9.4.3, and 4.1.1. In a letter dated November 15, 2006, the applicant submitted TR-30, "AP1000 CRDM Design," APP-GW-GLN-013, Revision 0 to provide the technical justification for the proposed changes.

As stated in Revision 15 to the AP1000 DCD, Sections 3.5.1.2.1.1, 3.9.4.1.1, and 4.1.1, the specified hydrostatic test pressure for the CRDM is 150 percent of the system design pressure. In Section 3.9.4.1.1 of the DCD, the attachment of the latch assembly housing is described as a shrink-fit and partial penetration weld of the latch assembly housing. However, the latch assembly housing will be welded to the CRDM nozzle by a bi-metallic weld. Also, Section 3.5.1.2.1.1 describes the attachment of the latch assembly housing to a head adapter when in fact the latch assembly housing will be welded to an Alloy 690 nozzle. In Revision 17 to the DCD, the applicant proposed to hydrostatically test the CRDM at 125 percent of system design pressure and to describe the correct fabrication sequence and terminology for the assembly.

In addition, in Sections 3.9.4.1.2, "Control Rod Withdrawal"; 3.9.4.1.3, "Control Rod Insertion"; and 3.9.4.1.4, "Holding and Tripping of the Control Rods," the applicant proposed modifications to the sequence of events for withdrawal, insertion, holding and tripping of control rods.

The DCD was modified to clarify the classification of the CRDM latch assembly, the CRDM drive rod assembly, CRDM coil stack assembly, and the CRDM position indicator.

3.9.4.1 Evaluation

3.9.4.1.1 Hydrostatic Testing and Attachment of the Latch Assembly Housing

The applicant revised AP1000 DCD, Sections 3.5.1.2.1.1, 3.9.4.1.1, and 4.1.1 to reduce the hydrostatic test pressure for the CRDM from 150 percent to 125 percent of system design pressure. The stated reason for this change was the requirements of the ASME Code. Section III, Paragraph NB-6221 specifies that nuclear power plant components are tested at 125 percent of system design pressure. The staff finds the proposed change acceptable because the proposed hydrostatic test pressure of 125 percent of system design pressure meets the requirements of ASME Code, Section III, which the NRC had incorporated by reference in 10 CFR 50.55a.

AP1000 DCD, Revision 15, Section 3.9.4.1.1, states the attachment of the latch assembly housing to the vessel head is accomplished by a shrink-fit and partial penetration weld. The applicant determined the latch housing will be welded to the Alloy 690 nozzle with a bi-metallic weld and the nozzle will be attached to the reactor vessel head by a shrink-fit and partial penetration weld. In Revision 19 to the DCD, the applicant revised Sections 3.5.1.2.1.1, 3.9.4.1.1, and 3.9.4.3 to describe the correct fabrication sequence and correct terminology for these components. The staff finds the proposed changes are an editorial change to the AP1000 DCD and as such do not affect the design basis of the component. Furthermore, the proposed changes describe the correct fabrication sequence, uses the correct terminology and, therefore, are acceptable.

The staff reviewed the proposed changes as they relate to Revision 17 to the AP1000 DCD. The proposed changes, as identified in TR-30, have been adequately incorporated into Revision 19 to the DCD. Accordingly, these changes are generic and are expected to be used by all COL applications referencing the AP1000 certified design.

3.9.4.1.2 Control Rod Sequence of Events

In Sections 3.9.4.1.2, “Control Rod Withdrawal,” 3.9.4.1.3, “Control Rod Insertion,” and 3.9.4.1.4, “Holding and Tripping of the Control Rods,” the applicant proposed to modify the control rod withdrawal and insertion sequence order. Specifically, during control rod withdrawal the moveable gripper coil B is in the de-energized (“OFF”) state instead of the energized (“ON”) state. Furthermore, insertion of control rods initiates with the moveable gripper coil B in the de-energized (“OFF”) state instead of the lift coil C in the energized (“ON”) state.

The applicant proposed to change the DCD in Section 3.9.4.1.4 “Holding and Tripping of the Control Rods” to be in accord with the proposed change in Section 3.9.4.1.1 “Control Rod Drive Mechanism (CRDM).” The proposed change reiterates that in the holding mode both the stationary gripper coil A and the moveable gripper coil B are energized. Additionally the applicant elaborates the drive rod assembly is held in position by three latches on the stationary gripper and three latches on the moveable gripper. As a result of the proposed modification, the applicant clarifies a reactor trip occurs when power to the stationary as well as to the moveable gripper coils is cut off.

The staff finds the proposed changes to the sequence of events for control rod withdrawal, control rod insertion, and holding and tripping the control rod do not adversely affect the ability of the AP1000 CRDM to perform its safety-related functions.

3.9.4.1.3 Seismic Qualification of CRDM

The staff became aware of discussions internationally concerning the classification and qualification of the CRDM latch assembly. Based on these discussions, the staff determined the seismic qualification of the CRDM standard design may not meet requirements. GDC 2 of Appendix A to 10 CFR Part 50 states that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, combined with appropriate effects of normal and accident conditions, without loss of capability to perform their safety functions.

In RAI-SRP3.9.4-EMB1-01, the staff requested that the applicant provide a justification to explain why the latch mechanism and coil stack assembly do not need to be seismically qualified to comply with GDC 2, or to revise the seismic classifications of the CRDM components to ensure adequate seismic qualification for the safety functions of the control rod drive system. In RAI-SRP3.9.4-EMB1-02, the staff requested further clarification on design changes discussed internationally.

The applicant provided justification for the equipment classification for the latch assembly and coil stack assembly and why they do not need to be seismically qualified. The justification is based on: 1) the design finality of the AP1000 DC; 2) the precedent of operating plants; and 3) the function of the latch assembly and coil stack assembly in the AP1000 CRDM.

The staff finds the applicant's justification for not qualifying the latch assembly using the precedent of operating plants and any postulated failure of the latch assembly results in a dropped rod and a subsequent increase in negative reactivity (justifications 2 and 3 above) unacceptable. Operating plants were licensed using a lower RRS compared to new reactors. The RRS for new reactors is much higher. Jamming of the latch mechanism is a postulated failure which results in no dropped rod and subsequently no reactivity change. Further, in the response to RAI-SRP3.9.4-EMB1-02, the applicant indicated that although international approaches to safety classification and requirements for safety class equipment may differ from the NRC requirements, it expects the design, fabrication and quality assurance requirements for the CRDM latch assemblies will remain common with the requirements for latch assemblies manufactured for U.S. applications.

In its response, the applicant referred to discussion in Chapter 15. There are only three postulated events that assume credit for reactivity control systems, other than a reactor trip to render the plant subcritical. These events are the steam-line break, feedwater-line break, and small break loss of coolant accident. The reactivity control systems in these accidents are the reactor trip system and the PXS. The probability of a common mode failure impairing the ability of the reactor trip system to perform its safety-related function is extremely low. However, analyses performed to demonstrate compliance with the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," demonstrate safety criteria would not be exceeded even if the control rod drive system were rendered incapable of functioning during anticipated transients for which its function would normally be expected. The evaluation demonstrates borated water from the CMT shuts down the reactor with no rods required, and the passive residual heat removal system provides sufficient core heat removal. Due to these additional safety measures, the applicant concluded the latch assembly and all other active mechanical components of the CRDM are not required to be classified as safety-related.

Based on finality, the low probability of common mode failure, and the argument that existing additional safety measures limit the safety consequence, the applicant has provided adequate justification to maintain the current classifications for the latch mechanism and coil stack assembly. Additionally, the applicant does not expect any changes to the design, fabrication and quality assurance requirements for the CRDM latch assemblies. The staff finds the responses to RAI-SRP3.9.4-EMB1-01 and RAI-SRP3.9.4-EMB1-02 acceptable and RAI-SRP-3.9.4-EMB1-02 is closed.

As a result of RAI-SRP3.9.4-EMB1-01, the applicant proposed modifications to DCD Tier 2 Table 3.2-3 to clarify the classification of the CRDM latch assembly, the CRDM drive rod assembly, CRDM coil stack assembly, and the CRDM position indicator. Additionally, DCD Tier 2, Section SR 3.1.4.3 (of the Chapter 16 TS) will be modified to include drop tests after each earthquake requiring shutdown.

The staff finds these proposed revisions acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.9.4.2 Conclusion

The staff further concludes the applicant's proposed changes do not adversely affect the ability of the AP1000 CRDM to perform its safety-related functions. On the basis that the AP1000 control rod drive system design continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds the changes to the CRDM design description provided in AP1000 DCD are acceptable.

3.9.5 Reactor Pressure Vessel Internals

In Revision 16 of the AP1000 DCD, the applicant added two new components (neutron panels to Section 3.9.5.1.1, and a flow skirt to Section 3.9.5.1.4) to the design of the reactor vessel internal structure. The subsequent Revision 17 of DCD Section 3.9.5 included minor changes to incorporate responses to the staff's RAIs for DCD Revision 16. DCD Revision 17 did not propose any additional new core support structure or reactor internals components requiring further technical evaluation.

3.9.5.1 Evaluation

The staff reviewed the proposed changes to the reactor vessel internals in the AP1000 DCD Revision 17 in accordance with the guidance in the NUREG-0800 Section 3.9.5, "Reactor Pressure Vessel Internals." The regulatory basis for Section 3.9.5 of the AP1000 DCD is documented in NUREG-1793. The following evaluation discusses the results of the staff's review.

3.9.5.1.1 Neutron Panels

In response to RAI-SRP3.9.5-EMB1-04 received in a letter dated June 20, 2008, the applicant stated the function of the neutron panels is to protect the reactor vessel from detrimental radiation effects by limiting the total exposure in the localized regions of the vessel wall in closest proximity to the core outer boundaries. The applicant also clarified the neutron panels are classified as internal structures, and, for conservatism, the neutron panels are analyzed in accordance with the requirements of ASME Code, Section III, Subsection NG. The neutron panels are fabricated from material complying with ASME III, NG-2000 and are designed and analyzed per ASME Code, Section III, NG-3000. The neutron panels are attached to the core barrel with threaded fasteners. The applicant also stated that the neutron panels have been sized to prevent excessive thermal loading on the bolts and to withstand flow, thermal and vibratory loading. In addition the bolts and preload of the bolts have been sized to accommodate radiation relaxation and radiation induced gamma heating such that the preload is maintained. These bolts are secured by locking devices. Oscillatory forces on the neutron panels have been calculated based on the turbulence in the annulus between the neutron panels and the reactor vessel based on the correlation with past scale model tests and CFD analysis. The analysis of the forces, as discussed was evaluated to assure the preload is maintained and design limits are achieved. The applicant also stated the AP1000 reactor vessel inside diameter has been increased by two inches over the core elevations where the neutron panels were added. This results in a net flow area increase of 4 percent in the downcomer relative to the flow area before the panels were added. Thus, the lower average downcomer velocity is expected to mitigate the potential for any adverse effects of flow-induced vibration caused by the added neutron panels.

3.9.5.1.2 Flow Skirt

The flow skirt is a perforated cylindrical ring structure attached to the reactor vessel bottom head at an elevation just below the LCSP. The flow skirt provides a more uniform distribution of inlet flow from the reactor vessel downcomer annulus to the core inlet nozzles in the LCSP. Although the flow skirt is welded to the reactor vessel, since the structure is located entirely within the pressure boundary, it is treated in the DCD as a reactor vessel internal structure. In response to RAI-SRP3.9.5-EMB1-01 received in a letter dated June 20, 2008, the applicant clarified although classified as an internal structure (as opposed to a core support structure), for conservatism the flow skirt is analyzed in accordance with the requirements of ASME Code, Section III, Subsection NG. The ASME Code jurisdictional boundary requires the attachment weld between the flow skirt and the reactor vessel flow skirt support lug to be designed and analyzed to ASME Code, Section III, Subsection NB-3200. All other design details of the flow skirt conform to ASME Code, Section III, Subsection NG-3000 requirements. The applicant also stated the flow skirt design includes flow-induced vibratory loading considerations including downcomer flow turbulence, random turbulence within the reactor vessel lower head, and vortex shedding through the flow skirt perforations. The flow skirt design specification requires the structural design qualification calculations for the flow skirt meet the requirements of ASME Code, Section III, Subsection NG-3000.

In response to RAI-SRP3.9.5-EMB1-02, dated June 20, 2008, the applicant stated the primary function of the flow skirt is to assure the distribution of flow entering the core is within prescribed limits for fuel assembly inlet flow mismatch. A CFD analysis of a reactor vessel/internals model, which included the inlet nozzle, downcomer, lower plenum (including secondary core support and vortex suppression structures), and LCSP, was performed by the applicant to determine the core inlet flow distribution. The CFD approach used in the analyses was used for analyses of similar operating reactor vessel internals geometry, and was benchmarked to scale model testing data with good agreement. The applicant performed analyses both with and without a flow skirt. Without the flow skirt the limits for uniformity of core inlet flow distribution were not met. In response to RAI-SRP3.9.5-EMB1-03, dated June 20, 2008, the applicant provided a figure of the flow skirt which clarified its form and function, and the staff considers this RAI closed.

3.9.5.1.3 Component Classification and Design Basis

The neutron panels are also classified as reactor internal structures (as opposed to core support structures), and, for conservatism, are designed according to the requirements of ASME III, Subsection NG-3000, even though ASME Code, Section III requires this approach only for internals components classified as core support structures. The flow skirt is also designed per the requirements of ASME Code, Section III, Subsection NG. As provided by ASME Code, Section III, Subsection NG-1122(b) and (c), these internal structure components must be constructed so as not to adversely affect the integrity of the core support structures, but the specific design requirements of ASME Code, Section III, Subsection NG are not required unless so stipulated by the designer. The applicant has conservatively chosen to use the requirements of Subsection NG-3000 for the design of both the flow skirt and the neutron panels.

The staff conducted a design audit at the applicant’s Energy Center in Monroeville, Pennsylvania during October 13-17, 2008. The audit included review of the ASME Code, Section III design documentation for the AP1000 RPV and the reactor internals and core support structure. The results of this audit are in an NRC letter dated December 30, 2008, Docket No. 52-006, Subject: Summary of the October 13-17, 2008, On-site Review of the AP1000 Component Design. The staff confirmed the neutron panels are part of the reactor internals design specification and design report, and the flow skirt has its separate design specification and analysis report. The audit verified the design bases for the neutron panels and flow skirt incorporate the requirements of ASME Code, Section III, Subsection NG-3000. The design analyses for the neutron panels show the results meet the design margins required by ASME Code, Section III, Subsection NG. Although the design report analysis for the flow skirt was not complete at the time of the audit, the flow skirt design specification clearly established the design requirements according to the provisions of ASME Code, Section III, Subsection NG. Therefore, the staff concluded the design methodology meets the review criteria of NUREG-0800 Section 3.9.5, and is acceptable.

In DCD Section 3.9.2.3, the applicant stated the results of the Trojan 1 reactor tests showed the lower internals vibrations are lower with neutron panels than with a circular thermal shield. Additionally, as stated above, a net flow area increase of 4 percent in the AP1000 downcomer relative to the flow area before the neutron panels were added results in a lower average flow velocity in the downcomer annulus. The lower average downcomer flow velocity will tend to mitigate the potential effects of any localized turbulence added by the neutron panels. On this basis, the staff concluded there is reasonable assurance the added neutron panels will not be adversely affected by FIV.

As indicated above, the applicant considered the flow-induced vibratory loading including downcomer flow turbulence and random turbulence for the flow skirt. The structural qualification requirements for the flow skirt and the neutron panels are consistent with the provisions of ASME Code, Section III, Subsection NG. The applicant's CFD analyses used for prediction of flow-induced vibratory loading coupled with pre-operational FIV testing (as discussed in Section 3.9.2.3 of NUREG-1793) will ensure there are no adverse effects of FIV and flow-excited acoustic resonances on the reactor vessel internal structures. On this basis, the staff finds the flow skirt and neutron panels will not cause adverse flow effects within the reactor vessel internal structures during normal operation or anticipated operational transients.

3.9.5.2 Conclusion

The applicant has met the regulatory requirements of GDC 1 and 10 CFR 50.55a by designing the neutron panels and the flow skirt to quality standards commensurate with the importance of the safety functions performed. The design criteria used for these two newly added reactor internals components are in compliance with the requirements of the 1998 Edition, including 1999 and 2000 Addenda, of ASME Code, Section III, Subsection NG-3000.

The applicant has met the regulatory requirements of GDC 2; GDC 4; and GDC 10, "Reactor Design," by designing these reactors internals components to withstand the effects of normal operation and postulated accident loadings with sufficient margin to maintain their structural integrity to assure they do not adversely affect the integrity of the safety-related reactor core support structures. The applicant has also designed these reactor internals components to assure acceptable fuel design and performance limits are met during conditions of normal operation and anticipated operational occurrences.

The staff concludes the design bases for the neutron panels and for the flow skirt meet the staff review criteria of NUREG-0800 Section 3.9.5, including the regulatory requirements of 10 CFR 50.55a, GDCs 1, 2, 4, and 10, and are, therefore, acceptable.

3.9.6 Testing of Pumps and Valves

In Revision 16 to the AP1000 DCD Tier 2, the applicant modified Section 3.9.6, "Inservice Testing of Pumps and Valves," including Table 3.9-16, "Valve Inservice Test Requirements." The applicant incorporated changes to AP1000 DCD Tier 2, Section 3.9.6, to support the description of the Inservice Testing (IST) Program required to be provided by a COL applicant.

3.9.6.1 Evaluation

In Section 3.9.6, "Testing of Pumps and Valves," of NUREG-1793, the NRC described its review of the description of the IST Program for the AP1000 design provided in AP1000 DCD Tier 2, Section 3.9.6. Other sections of the AP1000 DCD addressed the design of safety-related valves, and inservice inspection and testing of dynamic restraints. As discussed in NUREG-1793, the development of a complete plant-specific IST Program falls outside the scope of DC. At the DC stage, it is necessary to establish a baseline Code Edition and Addenda to ensure the IST requirements of the baseline ASME Code can be performed without exception, and that the design of the AP1000 systems and components provides access to permit the performance of testing pursuant to the NRC regulations specified in 10 CFR 50.55a.

AP1000 DCD Tier 2, Section 3.9.6 states inservice testing of ASME BPV Code, Section III, Class 1, 2 and 3 pumps and valves is performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable addenda, as required by 10 CFR 50.55a(f), except where specific relief has been granted by the NRC. The baseline ASME OM Code used to develop the IST plan for the AP1000 DC was the 1995 Edition and 1996 Addenda. AP1000 DCD Tier 2, Section 3.9.6 provides a general description of the IST Program to be developed for the AP1000 reactor to satisfy the requirements in 10 CFR 50.55a and the provisions of the ASME OM Code incorporated by reference in the NRC regulations. In NUREG-1793, the NRC found the IST Program description in the AP1000 DCD to be acceptable for the AP1000 DC, and that the AP1000 DCD had not taken exception to any ASME OM Code requirements established in the 1995 Edition and 1996 Addenda.

Since the issuance of NUREG-1793, the NRC has determined a COL applicant referencing the AP1000 design needs to fully describe the IST, motor-operated valve (MOV) testing and other operational programs as defined in Commission Paper SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." RG 1.206 provides guidance for a COL applicant in preparing and submitting its COL application in accordance with the NRC regulations. For example, Section C.IV.4 in RG 1.206 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs need to be included in the FSAR in a COL application to support a reasonable assurance finding of acceptability. A COL applicant may rely on information in the applicable DC to help provide a full description of the operational programs for the COL application. At a public meeting on March 26 and 27, 2008, the applicant indicated the AP1000 DCD will address issues common to COL applicants implementing the AP1000 design. Therefore, the staff reviewed the revision to the AP1000 DCD related to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints, including DCD provisions intended to minimize the supplemental information necessary to be provided by a

COL applicant in fully describing the operational programs in support of its COL application for an AP1000 reactor. As described below, the staff concludes the revision to Section 3.9.6 of the AP1000 DCD continues to provide an acceptable description of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints sufficient for the AP1000 DC in accordance with the NRC regulations and the ASME Code requirements incorporated by reference in the NRC regulations, with provisions for the consideration of lessons learned from nuclear power plant operating experience.

A COL applicant may reference the provisions in Section 3.9.6 to the AP1000 DCD as part of its responsibility to fully describe the IST, MOV testing, and other operational programs in support of its COL application. AP1000 DCD Tier 2, Section 3.9.6 states Table 3.9-16, “Valve Inservice Test Requirements,” identifies the components subject to the preservice and IST programs, and the method and frequency of preservice and inservice testing. The staff will evaluate the full description of the IST Program provided by a COL applicant during review of the COL application consistent with RG 1.206 and NUREG-0800 Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.” NUREG-1482, “Guidelines for Inservice Testing at Nuclear Power Plants,” provides guidance for the preparation of IST Program documentation and tables. Following COL issuance, the NRC will evaluate development and implementation of the IST Program prior to and during plant operation.

AP1000 DCD Tier 2, Section 3.9.6.1, “Inservice Testing of Pumps,” specifies the AP1000 reactor design does not include pumps with safety functions with the exception of the coastdown of the RCPs. The proposed changes to the AP1000 DCD do not affect the use of pumps with respect to safety-related applications. Therefore, the IST Program described in the proposed revision to the AP1000 DCD does not include pumps. As determined in NUREG-1793, the NRC considers the IST Program scope for the AP1000 design with respect to pumps to be acceptable.

AP1000 DCD Tier 2 discusses the functional design and qualification of safety-related valves and dynamic restraints in several sections. For example, Section 3.9.3.2, “Pump and Valve Operability Assurance,” in AP1000 DCD Tier 2, Chapter 3, “Design of Structures, Components, Equipment and Systems,” refers to operational tests to verify the valve opens and closes prior to installation. AP1000 DCD Tier 2, Section 3.9.3.2.2 specifies cold hydro tests, HFTs, periodic inservice inspections, and periodic inservice operations to be performed in situ to verify the functional capability of the valves. Section 5.4.8, “Valves,” of Section 5.4, “Component and Subsystem Design,” in AP1000 DCD Tier 2, Chapter 5, “Reactor Coolant System and Connected Systems,” includes provisions regarding design and qualification, and preoperational testing of valves within the scope of Chapter 5, and refers to these activities for other safety-related valves. AP1000 DCD Tier 2, Section 5.4.8.3, “Design Evaluation,” states the requirements for qualification testing of power-operated active valves are based on ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” as listed in AP1000 DCD Tier 2, Section 5.4.16, “References.” AP1000 DCD Tier 2, Section 5.4.9, “Reactor Coolant System Pressure Relief Devices,” includes provisions for design, testing, and inspection of relief devices in the RCS. AP1000 DCD Tier 2, Section 5.4.10, “Component Supports,” includes provisions for design, testing, and inspection of component supports in the RCS. During the public meeting on March 26 and 27, 2008, the applicant discussed its development of design and procurement specifications for safety-related valves and dynamic restraints for the AP1000 reactor design. In RAI-SRP3.9.6-CIB1-01, the staff requested that the applicant provide a schedule for the availability of the design and procurement specifications for safety-related valves and dynamic restraints to be used in the

AP1000 reactor for NRC review. In its response to this RAI, in a letter dated July 18, 2008, the applicant reported the design and procurement specifications would be made available for NRC review.

On October 14 and 15, 2008, the staff conducted an audit of design and procurement specifications for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the applicant's office in Monroeville, Pennsylvania. The staff found the applicant had included ASME Standard QME-1-2007 in its design and procurement specifications for AP1000 components. ASME QME-1-2007 incorporates lessons learned from valve testing and research programs performed by the nuclear industry and NRC Office of Nuclear Regulatory Research. In a memorandum dated November 6, 2008, the staff documented the results of the audit with the specific open items. The audit response was tracked as OI-SRP3.9.6-CIB1-01. In a letter dated January 26, 2010, the applicant provided its planned response to the audit follow-up items. First, the applicant stated a reference to ASME QME-1-2007 will be included in AP1000 DCD Tier 2, Section 3.9. Second, the applicant stated the basis for the assumptions for valve seat coefficients of friction for gate and globe valves is derived from the Joint Owners Group (JOG) Program on MOV periodic verification as a starting point for the initial actuator sizing.

The applicant indicated the final basis for the friction coefficient values will be derived in accordance with an approved methodology contained in ASME QME-1-2007. Third, the applicant stated the applicable valve design specification indicates active valves must be qualified in accordance with the ASME QME-1 standard, and the specification will be further clarified to indicate any existing testing used to demonstrate functional qualification must fully satisfy the provisions of ASME QME-1-2007. Fourth, the applicant stated the AP1000 DCD Tier 2, Figure 6.3-1, "Passive Core Cooling System Piping and Instrumentation Diagram," will be revised to include test connections to allow flow testing of Core Makeup Tank Discharge Check Valves PXS-V016A/B and V017A/B in both the forward and reverse directions. In September 2009, the NRC issued Revision 3 to RG 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," which accepts the use of ASME QME-1- 2007, with certain staff positions, for the functional design and qualification of safety-related pumps, valves, and dynamic restraints.

On May 17, 2010, the staff conducted a follow-up audit at the applicant's office in Rockville, Maryland, to review the revisions to the design and procurement specifications prepared since the October 2008 audit. The staff conducted telephone conferences on May 19 and 28, and June 10, 2010, to support close-out of the audit. Based on the follow-up audit, the staff found the applicant had updated the design and procurement specifications to address NRC comments provided during the October 2008 audit. For example, the staff found the design and procurement specifications require the application of ASME Standard QME-1-2007 for the qualification of mechanical equipment to be used in an AP1000 reactor.

The design and procurement specifications also had been revised to incorporate additional comments provided by the staff during the October 2008 audit. In a memorandum dated June 15, 2010, the staff documented the results of the follow-up audit. Based on the results of the follow-up audit, OI-SRP3.9.6-CIB1-01 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Revision 18 to the AP1000 DCD Tier 2 modified Section 6.3.2.2.8.1, "Manual Globe, Gate, and Check Valves," in Section 6.3, "Passive Core Cooling System," to remove the specification of check valves in the PXS as either piston type or swing type. In a letter dated May 21, 2010, under Change Number 4, the applicant indicated that core makeup tank discharge check valves PXS-PL-016A/B and PXS-PL-017A/B would be designed as in-line nozzle check valves rather than swing check valves as previously specified. The staff reviewed the design change for check valves PXS-PL-016A/B and PXS-PL-017A/B. The staff notes that AP1000 DCD Tier 2, Section 3.9.3.2.2, "Valve Operability," specifies that the qualification of functional capability of active valve assemblies will be performed in accordance with the requirements of ASME QME-1-2007. As noted above, the NRC staff has accepted the application of ASME QME-1-2007 in Revision 3 to RG 1.100 with certain conditions. Further, AP1000 DCD Tier 1, Table 2.2.3-1 lists check valves PXS-PL-016A/B and PXS-PL-017A/B as components to be addressed by the ITAAC specified in AP1000 DCD Tier 1, Table 2.2.3-4. Therefore, the NRC staff finds the design change for check valves PXS-PL-016A/B and PXS-PL-017A/B to be acceptable.

AP1000 DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis," describes tests to confirm piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state FIV and anticipated operational transient conditions. Section 14.2.9.1.7, "Expansion, Vibration and Dynamic Effects Testing," in AP1000 DCD Tier 2, Chapter 14, "Initial Test Program," states the purpose of the expansion, vibration and dynamic effects testing is to verify the safety-related, high energy piping and components are properly installed and supported such that, in addition to other factors, vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems. Nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance on reactor coolant, steam, and feedwater systems. As part of the functional design and qualification for AP1000 components, the COL applicant will be responsible for addressing the provisions in the AP1000 DCD for consideration of potential adverse flow effects on safety related valves and dynamic restraints within the IST Program in the reactor coolant, steam, and feedwater systems from hydraulic loading and acoustic resonance during plant operation.

AP1000 DCD Tier 2, Section 3.9.6.2, "Inservice Testing of Valves," refers to the use of nonintrusive techniques to periodically assess degradation and performance of selected valves. In RAI-SRP3.9.6-CIB1-02, the staff requested that the applicant to clarify the use of nonintrusive techniques within the IST Program to support implementation of this section by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, dated September 9, 2008, the applicant stated it will be the responsibility of the licensee to define the nonintrusive technique and methods for periodic assessment of check valve performance and degradation. Also in response to this RAI, the applicant modified Section 3.9.6.2 in Revision 17 to the AP1000 DCD Tier 2 to state inservice testing may incorporate the use of nonintrusive techniques to periodically assess degradation and performance of selected check valves. The staff finds that the applicant's response to this RAI and Revision 17 to the AP1000 DCD clarify the use of nonintrusive techniques referenced in the AP1000 DCD, and the COL licensee will define any nonintrusive techniques that will be implemented. Therefore, RAI-SRP3.9.6-CIB1-02 is closed.

The revision to AP1000 DCD Tier 2, Section 3.9.6.2 specifies testing of power-operated valves (POVs) used in the AP1000 reactor will utilize guidance from Generic Letter (GL) 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," and the JOG MOV Periodic Verification Program. The staff accepted the JOG Program on MOV periodic verification as an industry-wide response to GL 96-05 for valve age-related degradation in a safety evaluation dated September 25, 2006, and with a supplement dated September 18, 2008. In RAI-SRP3.9.6-CIB1-03, the staff requested that the applicant describe the incorporation of lessons learned from valve programs in planning the IST program for POVs other than MOVs to support implementation of this section by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008, the applicant stated AP1000 DCD Tier 2, Section 3.9.6.2 would be revised to address this RAI. As a result, Revision 17 to AP1000 DCD Tier 2, Section 3.9.6.2, states that guidance from applicable NRC generic letters and industry guidelines is reflected in the IST provisions in AP1000 DCD Tier 2, Table 3.9-16. Revision 17 to the AP1000 DCD also specifies lessons learned from GL 96-05 and the JOG MOV Periodic Verification Program are reflected in the IST Program and valve procurement testing requirements. Revision 17 to the AP1000 DCD indicates the IST Program requires periodic updating that takes into account changes to the diagnostic methods and test equipment, emergent industry issues, and equipment alignment. The staff finds the applicant response to RAI-SRP3.9.6-CIB1-03 and the provisions specified in Revision 17 to the AP1000 DCD provide an acceptable clarification as part of the AP1000 Design Certification that the lessons learned from valve operating experience and testing programs will be included in the IST and procurement programs for AP1000 nuclear power plants. RAISRP3.9.6-CIB1-03 is closed.

AP1000 DCD Tier 2, Section 3.9.6.2 states the operability test for safety-related POVs with an active function may be either a static or a dynamic (flow and differential pressure) test. In RAI-SRP3.9.6-CIB1-04, the staff requested that the applicant clarify the use of static tests for operability determinations of POVs to support implementation of this subsection by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008, the applicant stated AP1000 DCD Tier 2, Section 3.9.6.2 would be revised to address this RAI. As a result, Revision 17 to the AP1000 DCD Tier 2, Section 3.9.6.2 references Section 3.9.6.2.2 for the use of static or dynamic testing for safety-related POVs. The staff considers this clarification of Section 3.9.6.2 to be sufficient to close this RAI, but the use of static or dynamic testing for safety-related POVs will be addressed as part of RAI-SRP3.9.6-CIB1-08 discussed later in this safety evaluation. RAI-SRP3.9.6-CIB1-04 is closed.

The revision to AP1000 DCD Tier 2, Section 3.9.6.2.2 states the frequency for a position indication test will be once every 2 years unless otherwise justified. In RAI-SRP3.9.6-CIB1-07, the staff requested that the applicant clarify the need for a COL applicant to request relief from or an alternative to the ASME OM Code testing requirement with respect to position indication if the Code provisions are not satisfied. In its response to this RAI in a letter dated July 14, 2008, the applicant noted that AP1000 valves that require position indication testing, as documented in AP1000 DCD Tier 2, Table 3.9-16, are identified as having a 2 year frequency. The applicant indicated no relief is requested for position indication testing. The staff considers the position indication testing frequency in the AP1000 DCD to be consistent with the ASME OM Code. The COL applicant will need to request relief from, or an alternative to, the ASME OM Code provisions if the position indication testing frequency will not be satisfied. RAI-SRP3.9.6-CIB1-07 is resolved.

AP1000 DCD Tier 2, Section 3.9.6.2.2 discusses POV testing in a subsection titled "Power-Operated Valve Operability Tests." The revision to the AP1000 DCD specifies operability testing as required by 10 CFR 50.55a(b)(3)(ii) is performed on MOVs that are included in the ASME OM Code IST Program to demonstrate the MOVs are capable of performing their design-basis safety functions. In RAI-SRP3.9.6-CIB1-08, the staff requested that the applicant clarify the discussion of POV operability testing in the AP1000 DCD to support implementation of the DCD provisions by a COL applicant referencing the AP1000 reactor design. In response to this RAI, in a letter dated September 9, 2008, the applicant described planned changes to AP1000 DCD Tier 2, Section 3.9.6.2.2 to address this RAI. The staff determined RAI-SRP3.9.6-CIB1-08 needed to remain open until several aspects of the planned AP1000 DCD changes were clarified as discussed below for OI-SER3.9.6-CIB1-02, OI-SER3.9.6-CIB1-03, OI-SER3.9.6-CIB1-04, and OI-SER3.9.6-CIB1-05.

In OI-SRP3.9.6-CIB1-02, the staff tracked the need for the reference to static testing of valves in the AP1000 DCD to be consistent with the JOG MOV Periodic Verification Program, which might require dynamic testing based on the results of the evaluation of the MOV margin. In letters dated January 26, February 18, and March 5, 2010, the applicant provided planned changes to the AP1000 DCD Tier 2, Section 3.9.6 to specify POV testing will be consistent with the JOG MOV Periodic Verification Program, and removed the reference to static-only testing. The applicant also removed the discussion of testing of POVs outside the scope of the JOG MOV Periodic Verification Program because safety-related MOVs to be used at AP1000 plants will be within the scope of the JOG MOV Periodic Verification Program. The staff considers these planned DCD changes will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OI-SRP3.9.6-CIB1-02 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In OI-SRP3.9.6-CIB1-03, the staff tracked the need for the AP1000 DCD to specify the edition of the ASME Standard QME-1 referenced in Section 3.9 because the staff has not accepted ASME Standard QME-1 editions issued prior to 2007 as an acceptable functional qualification approach for valves. In letters dated January 26 and February 18, 2010, the applicant indicated AP1000 DCD Tier 2, Section 3.9 would be revised to reference ASME QME-1-2007. The staff considers this planned change to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In OI-SRP3.9.6-CIB1-04, the staff tracked the need for the planned application of ASME OM Code Case OMN-1, “Alternative Rules for Preservice and Inservice Testing of Certain Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants,” within the AP1000 IST Program to be consistent with the edition of Code Case OMN-1 accepted in RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” or to indicate the need for the submission of a request to implement an alternative to the ASME OM Code. In letters dated January 26, February 18, and March 5, 2010, the applicant provided a planned revision to AP1000 DCD Tier 2, Section 3.9.6 that will specify use of ASME OM Code cases must be consistent with RG 1.192. The staff considers this planned revision to the AP1000 DCD to be acceptable. A COL applicant or licensee planning to use an ASME OM Code case not accepted in RG 1.192 will need to submit a request to implement an alternative to ASME OM Code as required by 10 CFR 50.55a. The staff considers this planned change to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OISRP3.9.6-CIB1-04 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In OI-SRP3.9.6-CIB1-05, the staff tracked the need for the TS and TS Bases to be revised to be consistent with the ASME OM Code, such as in paragraph d of TS Section 5.5.3, and in References 4 and 5 to TS Bases for Surveillance Requirement 3.7.1.1. In its letter dated January 26, 2010, the applicant provided a planned revision to the AP1000 DCD TS and TS Bases to be consistent with the ASME OM Code. The staff considers these planned changes to the AP1000 DCD will resolve this portion of RAI-SRP3.9.6-CIB1-08. Therefore, OI-SRP3.9.6-CIB1-05 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

AP1000 DCD Tier 2, Section 3.9.6.2.2 discusses check valve testing in a subsection titled "Check Valve Exercise Tests." The revision to the AP1000 DCD Tier 2 indicates check valves must be exercised in the open and closed directions. In RAI-SRP3.9.6-CIB1-09, the staff requested that the applicant clarify the discussion of the AP1000 IST Program to support implementation of the AP1000 DCD provisions for check valves by a COL applicant referencing the AP1000 reactor design. In its response to this RAI, in a letter dated September 9, 2008, the applicant specified the acceptance criteria for assessing individual valve performance will be based on full open (full disk lift or achieving design accident flow rates) and valve closure verification using differential pressure/backflow tests. The applicant stated it is anticipated that Appendix II, "Check Valve Condition Monitoring Program," of the ASME OM Code will be implemented after sufficient operational data are obtained for the AP1000 check valves. The staff considered the RAI response to be acceptable, but the AP1000 DCD needed to include the specified acceptance criteria for check valve testing. The staff tracked this item as OI-3.9.6-CIB1-06. In letters dated January 26 and March 5, 2010, the applicant provided planned changes to AP1000 DCD Tier 2, Section 3.9.6 and Table 3.9-16 to include the check valve test acceptance criteria and to identify those check valves that will need to have a mechanical exerciser installed in lieu of flow testing. The staff considers that these planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-09. Therefore, OISRP3.9.6-CIB1-06 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The section titled "Pressure/Vacuum Relief Devices," in AP1000 DCD, Tier 2, Section 3.9.6.2.2 addresses the IST Program for pressure and vacuum relief devices. In RAI-SRP3.9.6-CIB1-10, the staff requested that the applicant provide additional information in specific areas regarding the IST Program for safety and relief valves. In response to this RAI, in a letter dated September 9, 2008, the applicant stated RCS pressure relief devices are discussed in AP1000 DCD Tier 2, Section 5.4.9. Pressure relief devices for other ASME Code systems are described with the applicable system in the AP1000 DCD. All safety and relief valves included in the IST Program will be tested to the rules of Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," of the ASME OM Code. ASME Code Class 1, 2, and 3 pressure relief valves are identified in AP1000 DCD Tier 2, Table 3.9-16. The staff considers this clarification of the applicable provisions for safety and relief valves to be consistent with the ASME OM Code. RAI-SRP3.9.6-CIB1-10 is resolved.

AP1000 DCD Tier 2, Table 3.9-16 lists the valves in the IST Program with their valve and actuator type, safety-related missions, safety functions, ASME Class and IST Category, and IST type and frequency. In RAI-SRP3.9.6-CIB1-11, the staff requested that the applicant update Note 31 of Table 3.9-16, which addresses operability testing of various POVs to reflect changes to the AP1000 DCD. In its response to this RAI, in a letter dated July 18, 2008, the applicant stated the MOV and air-operated valve (AOV) programs are expected to incorporate attributes for a successful POV periodic verification program as discussed in Regulatory Issue Summary (RIS) 2000-03, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves under Design Basis Conditions." The applicant provided a planned revision to Note 31 of Table 3.9-16 stating the applicable valves are subject to operability testing per the NRC regulations in 10 CFR 50.55a. The staff considered Note 31 needed to be clarified to be consistent with the JOG MOV Periodic Verification Program, and to include the expectation indicated by the applicant in the RAI response that the MOV and AOV programs will incorporate attributes for a successful POV periodic verification program as discussed in RIS 2000-03. The staff tracked this item as Open Item OI-SRP3.9.6-CIB1-07. In its letter dated January 26, 2010, the applicant provided a planned revision to Note 31 in Table 3.9-16, which will specify valve test frequencies will be established in accordance with the results of the JOG MOV Periodic Verification Program. The planned Note 31 revision will also state the JOG approach will be applied to all actuator types and the attributes of the POV programs will include lessons learned as delineated in RIS 2000-03. The staff considers these planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-11. Therefore, OI-SRP3.9.6-CIB1-07 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The revision to the AP1000 DCD includes changes to several notes in Table 3.9-16. In RAI-SRP3.9.6-CIB1-12, the staff requested that the applicant discuss the basis for the changes specified to Table 3.9-16. In its response to this RAI, in a letter dated September 9, 2008, the applicant stated Note 2 addressing valve safety functions includes such cases where normal valve operator action moves the valve to the open or closed position by de-energizing the operator electrically, by venting air, or both, then the exercise test will satisfy the fail-safe test requirements and an additional test for fail-safe testing will not be performed. Note 20 indicates the MSIVs and main feedwater isolation valves (MFIVs) will not be exercised during power operation to avoid a potential plant transient and reactor trip consistent with the guidance in NUREG-1482. Note 33 applies to fuel transfer tube isolation manual valve FHS-PL-V001 that will be tested consistent with 10 CFR 50.55a(b)(3)(vi) at a 2 year interval. Note 38 applies to main control room emergency habitability system (VES) pressure regulating valves that are exempt from the ASME OM Code, but the applicant stated it would revise the note in Table 3.9-16 to clarify the testing for these valves. As a result, Revision 17 to the AP1000 DCD, Section 3.9.6 modifies Note 38 to state exercise stroke tests for the VES pressure regulating valves will consist of a pressure drop test across the valve using the downstream test connection to ensure adequate testing of the valves. The staff finds the applicant’s response to RAI-SRP3.9.6-CIB1-12, and Revision 17 to the AP1000 DCD, Section 3.9.6 adequately clarify the testing for the valves described in the applicable notes in Table 3.9-16 discussed in this RAI to be consistent with the ASME OM Code and the NRC regulations. RAI-SRP3.9.6-CIB1-12 is closed.

The revision to the AP1000 DCD Tier 2 modifies Section 3.9.6.2.2 in a section titled “Remote Valve Position Indication Inservice Tests” to state position indication testing requirements for passive valves are identified in Table 3.9-16. In RAI-SRP3.9.6-CIB1-13, the staff requested that the applicant clarify this modification. In its response to this RAI, in a letter dated July 24, 2008, the applicant stated passive valves with remote position indication will be locally observed to verify the remote position indication accurately reflects valve position. All valves requiring position indication verification will be exercised during the position indication test such that the open and closed positions can be verified. The frequency of this test will be once every 2 years. All passive valves with test requirements are included in AP1000 DCD Tier 2, Table 3.9-16. The staff considers the incorporation of passive valves with test requirements in Table 3.9-16 to be consistent with the requirements of the ASME OM Code, Subsection ISTC-3700, “Position Verification Testing.” RAI-SRP3.9.6-CIB1-13 is resolved.

Section 3.9.6.2.2 of the AP1000 DCD Tier 2 under Manual/Power-Operated Valve Tests states the IST requirements for measuring stroke time for valves in AP1000 reactor will be completed in conjunction with a valve exercise test, and the stroke time test is not identified as a separate test. In RAI-SRP3.9.6-CIB1-14, the staff requested that the applicant clarify the stroke time testing provisions in the AP1000 DCD. In its response to this RAI, in a letter dated July 24, 2008, the applicant stated each POV is stroke-time tested when the full stroke exercise test is performed. The stroke time open or closed will match the safety-related mission (i.e., transfer open or closed) as identified in AP1000 DCD Tier 2, Table 3.9-16. The staff considers the IST description for stroke-time testing specified in Table 3.9-16 to be consistent with the ASME OM Code. RAI-SRP3.9.6-CIB1-14 is resolved.

Section 3.9.6.2.2 of the AP1000 DCD Tier 2 under Manual/Power-Operated Valve Tests states safety-related valves that fail to the safety-related actuation position to perform the safety-related missions are subject to a valve exercise inservice test. In RAI-SRP3.9.6-CIB1-15, the staff requested that the applicant clarify the discussion of fail safe testing. In its response to this RAI, in a letter dated July 24, 2008, the applicant stated the exercise test will satisfy the fail safe test requirements in cases where normal valve operator action moves the valve to the open or closed position by de-energizing the operator electrically, by venting air, or both. The applicant indicated remote position indication is used as applicable to verify proper fail safe operation, provided the indication system for the valve is periodically verified in accordance with ASME OM Code, Subsection ISTC-3700. The valves listed in Table 3.9-16 with an Active to Failed Safety Function are designed for only one safety-related mission direction with the fail position being the transfer open or transfer close position. The staff considered the reference to ASME OM Code, Subsection ISTC-3700, needed to be clarified to confirm the exercise test frequency requirements specified in the ASME OM Code for these valves will be satisfied. This item was tracked as Open Item OI-SRP3.9.6-CIB1-08. In its letter dated January 26, 2010, the applicant noted the Position Indication Verification Test is separate and independent of the Fail Safe Test. The applicant provided a planned revision to AP1000 DCD Tier 2, Table 3.9-16 to indicate a separate Fail Safe test for the applicable valves with fail safe functions. The staff considered these planned changes to the AP1000 DCD would resolve RAI-SRP3.9.6-CIB1-15. Therefore, OI-SRP3.9.6-CIB1-08 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The revision to Section 3.9.6.2.2 of the AP1000 DCD Tier 2 under Check Valve Exercise Tests states, if exercise testing during a refueling outage is not practical, then another method is applied, such as nonintrusive diagnostic techniques or valve disassembly and inspection. In RAI-SRP3.9.6-CIB1-16, the staff requested that the applicant clarify the revision to the AP1000 DCD for check valve exercise testing. In its response to this RAI, dated July 24, 2008, the applicant stated no check valves for which exercise tests are recommended have been identified, which cannot be full stroke exercised. As a result, neither nonintrusive techniques nor disassembly/inspection is required as part of the AP1000 certified design. If check valves are identified for which exercise tests are recommended but not practical due to operational issues or changes to the ASME OM Code, the applicant stated it will be the responsibility of the licensee to define the types of nonintrusive diagnostic techniques to be used. To clarify this provision, Revision 17 to AP1000 DCD Tier 2, Section 3.9.6.2.2, specifies the check valves included in the IST Program outlined in Table 3.9-16 do not require another means as an alternate to exercise testing based on the ASME OM Code used to develop the IST plan for the AP1000 Design Certification. The staff finds that the applicant’s response to RAI-SRP3.9.6-CIB1-16 and Revision 17 to the AP1000 DCD provide an acceptable clarification of the exercise testing for check valves. RAI-SRP3.9.6-CIB1-16 is closed.

Section 3.9.6.2.2 of the AP1000 DCD Tier 2 under Pressure/Vacuum Relief Devices states the frequency for this inservice test is every 5 years for ASME Class 1 and main steam safety valves, or every 10 years for ASME Classes 2 and 3 devices. The ASME OM Code also requires 20 percent of the valves from each valve group be tested within any 24 month interval for Class 1 and main steam safety valves, and within any 48 month interval for Class 2 and 3 devices. In RAI-SRP3.9.6-CIB1-18, the staff requested that the applicant discuss the requirement to test 20 percent of each valve group within the interval required by the ASME OM

Code. In response to this RAI, in a letter dated July 24, 2008, the applicant indicated that AP1000 DCD Tier 2, Table 3.9-16 includes the provision for 20 percent of the valves from each group to be tested. Further, Revision 17 to the AP1000 DCD Tier 2, Section 3.9.6.2.2, clarifies the provision that 20 percent for the valves from each group will be tested within any 24 month interval for Class 1 and main steam safety valves and within any 48 month interval for Class 2 and 3 devices. The staff finds the applicant's response to RAI-SRP3.9.6-CIB1-18 as incorporated into AP1000 DCD Revision 17 provides an acceptable clarification to ensure the IST activities are consistent with the ASME OM Code. RAI-SRP3.9.6-CIB1-18 is resolved.

The revision to the AP1000 DCD Tier 2 modifies Section 3.9.6.2.3 to state the sample disassembly examination program shall group check valves of similar design, application, and service condition, and shall require a periodic examination of one valve from each group. In RAI-SRP3.9.6-CIB1-19, the staff requested that the applicant clarify its plans for the disassembly examination program for check valves. In its response to this RAI, in a letter dated July 24, 2008, the applicant stated all check valves in the AP1000 IST Program outlined in AP1000 DCD Tier 2, Table 3.9-16 are capable of being full stroke exercise tested based on the ASME OM Code (1995 Edition and 1996 Addenda) used to develop the IST plan for the AP1000 Design Certification. The applicant indicated it will be the responsibility of the licensee to define requirements of a disassembly and inspection program if check valves are identified for which exercise tests are recommended, but are not practical due to operational issues or changes in the ASME OM Code. The provisions in the AP1000 DCD for check valve exercise tests are consistent with ASME OM Code and, therefore, are acceptable. RAI-SRP3.9.6-CIB1-19 is resolved.

The revision to AP1000 DCD Tier 2 modifies Table 3.9-16 to identify valve type, operator, class and category for valves in the AP1000 IST Program. In RAI-SRP3.9.6-CIB1-20, the staff requested that the applicant clarify several items in Table 3.9-16. In its response to this RAI, in a letter dated September 9, 2008, the applicant discussed each specific RAI item and planned changes to the AP1000 DCD. For example, the applicant provided a modification to Table 3.9-16 (incorporated in Revision 17 to the AP1000 DCD) that includes a provision for full stroke exercising during refueling outages for service air supply containment isolation valve CAS-PL-V205. The applicant stated that the chemical volume and control system (CVS) containment isolation valves CVS-PL-V045, CVS-PL-V047, CVS-PL-V090, CVS-PL-V091, CVSPL-V092, and CVS-PL-V094 will receive only a leakage test in accordance with 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” The applicant clarified that the air operator for RCS purification return line stop valve CVS-PL-V081 does not perform a safety function, and the valve will act as a simple check valve upon loss of power. The applicant provided a modification to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) that specifies full stroke exercise tests during refueling outages for demineralized water supply containment isolation check valve DWS-PL-V245 and fire water containment supply isolation check valve FPS-PL-V052.

The applicant provided a correction to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) to reflect the 2 year test frequency and IST Category C for ADS discharge header vacuum relief valve RCS-PL-V010A and V010B. The applicant clarified that the main control room emergency habitability system pressure regulating valves VES-PL-V002A and V002B are pressure regulating valves that are not part of the ASME OM Code IST Program, and provided a modification to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) to specify these valves are part of an augmented inspection program. The applicant stated Note 3 in Table 3.9-16 would be revised to remove the discussion of PRA for the ADS valves. The applicant stated the leak testing for valves CVS-PL-V001, V002, V080, V081, V082, V084, and V085 described in

Note 32 is beyond the ASME OM Code IST program, and is part of an augmented testing program. The applicant provided a modification to Table 3.9-16 (incorporated in AP1000 DCD, Revision 17) to correct the categorization of these CVS valves from Category A to Category B or C, which do not require OM Code leak testing.

The staff determined several aspects of this RAI response needed to be clarified and tracked this item as OI-SRP3.9.6-CIB1-09. The applicant addressed this open item in its letter dated January 26, 2010. First, the applicant stated CVS valves CVS-PL-V045, V047, V090, V091, V092, and V094 have a safety function to transfer closed for containment isolation and do not serve an RCS pressure boundary function. The applicant provided a planned revision to AP1000 DCD Tier 2, Table 3.9-16 to correct the function indication for these valves. Second, the applicant provided a planned revision to Note 3 in Table 3.9-16 to ensure consistency with RAI response. Third, the applicant clarified its response regarding the categorization of the CVS valves discussed in Note 32. The staff considers the clarifications and the planned changes to the AP1000 DCD will resolve RAI-SRP3.9.6-CIB1-20. Therefore, OI-SRP3.9.6-CIB1-09 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 3.9.6.2.2 of the AP1000 DCD Tier 2 under Check Valve Low Differential Pressure Tests identifies low differential pressure testing as an inservice test is performed in addition to exercise inservice tests once each refueling cycle. In RAI-SRP3.9.6-CIB1-17, the staff requested that the applicant clarify the discussion of low differential pressure testing. In its response to this RAI, in a letter dated July 24, 2008, the applicant stated the low differential pressure testing is part of an augmented test activity similar to that established for the AP600 reactor design during staff review of that design certification. As a result, Revision 17 to AP1000 DCD Tier 2, Section 3.9.6.2.2, indicates the low differential pressure testing is not required by the ASME OM Code, but is part of an augmented inspection program. In its RAI response, the applicant indicated AP1000 DCD Tier 2, Table 3.9-16 will be revised to specify this test will be performed once every refueling cycle. The staff finds the applicant's response to RAI-SRP3.9.6-CIB1-17 adequately clarifies the AP1000 test activities to be consistent with the AP600 certified design. However, the planned changes to Table 3.9-16 for the applicable check valves did not appear to be included in Revision 17 to the AP1000 DCD. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The revision to AP1000 DCD Tier 2 includes a new Section 3.9.3.4.4, "Inspection, Testing, Repair and/or Replacement of Snubbers," which specifies a program for inservice examination and testing of dynamic supports (snubbers) to be used in the AP1000 reactor will be prepared in accordance with the requirements of ASME OM Code, Subsection ISTD. AP1000 DCD Tier 2, Section 3.9.3.4.4 indicates details of the snubber inservice examination and testing program, including test schedules and frequencies, will be reported in the inservice inspection and testing plan included in the IST Program required by AP1000 DCD Tier 2, Section 3.9.8.3, "Snubber Operability Testing." AP1000 DCD Tier 2, Section 3.9.8.3 states a COL applicant referencing the AP1000 design will develop a program to verify operability of essential snubbers. The staff finds the provision in the AP1000 DCD for application of the ASME OM Code, Subsection ISTD, in the examination and testing of dynamic supports to be acceptable for the AP1000 Design Certification. The COL applicant will be responsible for satisfying the COL Information Item in AP1000 DCD Tier 2, Section 3.9.8.3.

The staff reviewed the revisions to the AP1000 DCD with respect to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to be used at an

AP1000 nuclear power plant. The staff finds the changes are generic and are expected to be applicable to all COL applications referencing the AP1000 certified design.

3.9.6.2 Conclusion

The staff concludes the revision to the AP1000 DCD continues to support the design aspects for the functional design, qualification, and IST programs for safety-related valves and dynamic restraints in the applicable NRC regulations for the AP1000 DC. The revision to the AP1000 certified design provides sufficient information to satisfy 10 CFR Part 50 and 10 CFR Part 52 for the design aspects of the functional design, qualification, and IST programs for safety-related valves and dynamic restraints to be used in the AP1000 reactor. The staff will review the operational program aspects regarding the functional design, qualification, and IST programs for safety-related valves and dynamic restraints in a COL application referencing the AP1000 certified design as part of the COL application review process.

3.9.7 Integrated Head Package

The integrated head package (IHP) provides the ability to rapidly disconnect cables including the CRDM power cables, digital rod position indication cables, and in-core instrument cables from the IHP components. The rapid disconnection of these cables provides the ability to move the IHP components as an assembly to permit the expedited lifting and removal of the reactor vessel head. In addition, the IHP provides support for the vessel head stud tensioner/detensioner during refueling. The IHP includes a lifting rig, seismic restraints for CRDMs, and support for the following IHP components: reactor head vent piping, cable bridge, power cables, cables and guide tubes for in-core instrumentation, cable supports, and shroud assembly.

By letter dated November 14, 2006, the applicant submitted TR-61, “AP1000 Integrated Head Package,” APP-GW-GLN-014. The purpose of TR-61 was to address changes in the IHP described in Revision 15 to the AP1000 DCD as reviewed by the staff in NUREG-1793.

Following a preliminary review, the staff requested additional information in a March 29, 2007 letter, via questions RAI-TR61-01 through RAI-TR61-04. By letter dated April 13, 2007, the applicant provided responses to the staff’s questions. It should be noted much of the staff’s focus in the review of TR-61 was associated with the change in the IHP design related to the removal of the CRDM cooling fans from the IHP to a separate structure and the resulting questions related to the adequacy of CRDM cooling.

The applicant subsequently submitted Revisions 16 and 17 to the DCD. In Section 3.9.7 of Revision 17 to the DCD, the applicant, again, proposes to attach the CRDM cooling fans to the IHP. In addition, the following changes are proposed:

- In the first paragraph of Section 3.9.7, the cable bridge is included in the IHP description but the guide tubes for in-core instrumentation are excluded.

- In Section 3.9.7.1, the shroud and CRDM seismic support plate, are no longer in the list of components which are required to provide seismic restraint for the CRDM and the valves and piping of the reactor head vent. The CRDM and the valves and piping of the reactor head vent still require seismic restraints. These components are AP1000 equipment Class C, seismic Category I and are designed in accordance with the ASME Code, Section III, Subsection NF requirements.

- The instrumentation guide tubes and the instrumentation support structure are excluded from those components function as part of the lifting rig and are required to be capable of lifting and carrying the total assembled load of the IHP.

- The components of the in-core instrumentation system (IIS) that interface with the IHP are the QuickLoc stalk assembly and the IIS cables and connectors. These have been excluded from the IHP description.

- The shroud assembly is required to provide radiation shielding of the CRDMs but the conduit for in-core instrumentation when the instrumentation is withdrawn into the conduit is not required to provide shielding. The radiation level at the exterior surface of the shroud during refueling with the in-core instrument thimble withdrawn is excluded from the discussion in the radiation levels discussed in Section 12.2.

- The description of the IHP in Section 3.9.7.2 excludes the In-core Instrumentation support structure.

- The description of the lifting system is modified. The lifting system attaches to the CRDM seismic support structure. The lift lugs transfer the head load during a head lift from the head attachment lugs; however, the attachment is no longer through the CRDM seismic support structure to the lift rig.

- In the description of the mechanism seismic support structure has been modified to reflect minor, proposed changes in the support structure.

- The description of the in-core instrumentation-support structure (IISS) has been changed to discuss the in-core instrumentation. The following statements related to the support structure have been deleted:

  – The in-core instrumentation support structure is used during refueling operations. This support structure is used for withdrawing the in-core instrumentation thimble assemblies into the integrated head package. It protects and supports the thimble assemblies when they are in the fully withdrawn position.

  – Also, the in-core instrumentation support structure includes a platform which provides access to the in-core instrumentation during maintenance and refueling and to attach the lifting system to the crane hook.

3.9.7.1 Evaluation

The staff reviewed the proposed changes related to Section 3.9.7 of AP1000 DCD Revision 17, including TR-61. The AP1000 IHP continues to meet all applicable acceptance criteria and requirements, as discussed below. The components of the IHP, which provide seismic support including the CRDM seismic support and the shroud, are designed using the ASME Code, Section III, Subsection NF which satisfies the limit on deflection of the top of the CRDM rod travel housing. The components of the IHP included in the load path of the lifting rig are designed to satisfy the requirements for lifting of heavy loads in NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” July 1980. The criteria of ANSI N14.6-1978, “Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," are used to evaluate the loads and stresses during a lift.

Those cables and connectors for the IIS are required to meet Class 1E requirements are evaluated for environmental conditions including normal operation and postulated accident conditions.

Components required to provide seismic restraint for the CRDMs and the valves and piping of the reactor head vent are AP1000 equipment Class C, seismic Category I and are designed in accordance with the ASME Code, Section III, Subsection NF requirements.

The loads and loading combinations due to seismic loads for these components are developed using the appropriate seismic spectra.

The structural design of the IHP is based on a design temperature consistent with the heat loads from the vessel head, the CRDMs, and electrical power cables. The design also considers changes in temperature resulting from plant design transients and loss of power to the cooling fans.

Components required to provide cooling to the CRDMs are non-safety-related AP1000 equipment Class E. Section 4.6 of the DCD Revision 17, offers a discussion of the effect of failure of cooling of the CRDMs.

Those components that function as part of the lifting rig are required to be capable of lifting and carrying the total assembled load of the IHP which includes the vessel head, CRDMs, CRDM seismic supports, shroud, cooling ducts, and insulation. The lifting rig components are required to meet the guidance for special lifting rigs, in NUREG-0612. The lifting rig components are non-safety-related, AP1000 equipment Class E.

The electrical cables and connectors, within the IHP, for the IIS are AP1000 equipment Class C, Class 1E. The other cables within the IHP, including power cables and cables for the digital rod position indicator system, are not Class 1E. The cable support provides seismic support and maintains separation for instrumentation and power cables.

3.9.7.2 Conclusion

The components of the IHP, which provide seismic support including the CRDM seismic support and the shroud, are designed using the ASME Code, Section III, Subsection NF. The IHP satisfies the limit on deflection of the top of the CRDM rod travel housing. The components of the IHP included in the load path of the lifting rig are designed to satisfy the requirements for lifting of heavy loads in NUREG-0612. The criteria of ANSI N14.6 are used to evaluate the loads and stresses during lifting.

Those cables and connectors for the IIS are required to meet Class 1E requirements are evaluated for environmental conditions including normal operation and postulated accident conditions. Accordingly, the staff concludes the AP1000 IHP design meets the requirements of 10 CFR Part 50, Appendix A, GDC 1; GDC 2; and GDC 30, "Quality of Reactor Coolant Pressure Boundary"; and 10 CFR Part 50, Appendix S; therefore, the proposed changes to Section 3.9.7 of the AP1000 DCD are acceptable.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

In Revision 17 of the DCD, Section 3.10, the applicant proposed some editorial and technical changes and clarifications. A summary of the major changes is described below.

One of the significant changes from DCD Revision 15 to DCD Revision 17 is that the applicant decided not to use Experience-Based Qualification Method for seismic qualification of AP1000 mechanical and electrical equipment. Therefore, all statements related to the experience-based qualification have been deleted or revised. For example, Section 3.10.6 and Item E.7 of Attachment E of Appendix D have been deleted.

In the introductory statements for Section 3.10 of AP1000 DCD Revision 17, a new paragraph was added to address the CSDRS exceedance in the high frequency spectrum region at some CEUS rock sites. A new Reference 3 was added to DCD Revision 17 and this new Reference 3 (Not "Reference 5" as indicated in the new paragraph) is related to the "AP1000 Design Control Document High Frequency Seismic Tier 1 Changes." The Tier 2 material related to the high frequency seismic input is provided in AP1000 DCD Revision 17, Appendix 3I.

Appendix 3I of AP1000 DCD addresses the effect of HRHF seismic input. The AP1000 HRHF evaluation study is reported in TR-115, which is referenced in AP1000 DCD Revision 17. In the course of reviewing TR-115, staff generated a list of RAIs, which is applicable to DCD Appendix 3I of AP1000 DCD Revisions 16 and 17.

3.10.1 Evaluation

The staff reviewed the major changes to Section 3.10 of the AP1000 DCD Revision 17 in accordance with the guidance in: (1) NUREG-0800 Section 3.10, "Seismic and Dynamic Qualification of Mechanical Electrical Equipment"; (2) DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications," May 19, 2008; and (3) SECY-93-087, including SRM93-087 issued on July 21, 1993. The regulatory basis for Section 3.10 of the AP1000 DCD is documented in NUREG-1793.

The changes in Appendix 3I related to Section 3.10 are mainly provided in Section 3I.6.4. The changes involve editorial clarifications and technical revisions. The results of the staff's review of the list of RAI responses are described below.

In RAI-SRP3.10-EMB-01, the staff requested that the applicant describe the screening process for potential high frequency sensitive mechanical and electrical equipment and components, and to provide a list of equipment including the justification for screening in or screening out. The detailed response to this RAI was initially submitted in a letter dated May 28, 2008, and later, a revised response was provided dated August 21, 2008. The applicant stated the AP1000 screening process for potential high frequency sensitive equipment is consistent with the NRC requirements in Section 4.0, "Identification and Evaluation of HF Sensitive Mechanical and Electrical Equipment/Components," of DC/COL-ISG-1, and guidelines identified in the EPRI White Paper, "Considerations for NPP Equipment and Structures Subjected to Response Levels Caused by High Frequency Ground Motions," transmitted to the NRC on March 16, 2007.

The goal of the AP1000 HRHF screening program is to identify those safety-related equipment and components are potentially HRHF-sensitive and show them to be acceptable for their specific application (screened-out). The AP1000 HRHF screening program is a two-step process; the first step is a HRHF susceptibility review to identify potential high frequency sensitive safety-related equipment. The second step is the screened-out equipment process to demonstrate its acceptability for the HRHF seismic excitation. Evaluation of screened-in equipment as defined in DC/COL-ISG-1 is not performed because all safety-related equipment is screened-in will be eliminated or shown to be acceptable through a design change process. Additional information is provided in Appendix 3I.6.4 of AP1000 DCD, Revision 17.

The staff reviewed the applicant's response related to the criteria and procedure for the AP1000 HRHF screening program as described above, and finds the response to be acceptable. The staff considers RAI-SRP3.10-EMB-01 to be closed.

In RAI-SRP3.10-EMB-02, the staff requested that the applicant explain, with respect to TR-115 Section 6.4.5, "Screening Process," its justification for using 50 Hz as the cut-off natural frequency for the Group No. 1 rugged equipment in the screening process, and to explain whether the electrical/electronic equipment/devices with natural frequencies greater than 50 Hz are considered as rugged equipment. The staff also requested that the applicant provide justification for not requiring additional evaluation for high frequency seismic inputs for equipment.

In Section 6.4.5 of TR-115 for the Screening Process, the applicant concluded safety-related equipment may be screened and grouped as follows: Group No.1 – Rugged equipment with dominant natural frequencies above 50 Hz; Group No. 2 – Cabinets and other equipment which exhibit dominant natural frequencies below HRHF exceedance range; and Group No. 3 – safety-related equipment which exhibit dominant natural frequencies in HRHF exceedance range. For Group No. 1 and Group No. 2 equipment, no additional evaluation for high frequency seismic input is necessary. For Group No. 3 equipment, the equipment will be subjected to supplemental high frequency seismic evaluation to verify acceptability.

In response to RAI-SRP3.10-EMB-02, dated May 28, 2008, the applicant stated that for AP1000, the frequency range of interest in the screening process is 25 Hz to 50 Hz. This range coincides with the peak region of the HRHF ground motion. Since the AP1000 plant building structure's dominant natural frequencies are considerably lower than 50 Hz, the horizontal and vertical GMRS above 50 Hz will not be amplified significantly and their response will dissipate quickly as it travels through the building structure. The worst case seismic loading will occur when the fundamental frequencies of the potential HRHF-sensitive equipment coincide with the peak of the response spectra. In addition, the applicant noted from review of AP1000 HRHF ISRS generated from the HRHF ground motions above 50 Hz, the ZPA regions of the response spectra are being approached. The applicant further stated equipment designs with dominant natural frequencies above 50 Hz are inherently rugged. The highly unlikely case of HRHF-sensitive equipment with a natural frequency of 55 Hz, for example, is a special class and would require combining screening process Groups Nos. 1 and 3. For this condition, the Group No. 3 process would govern and the equipment would be subjected to a supplemental HRHF seismic evaluation/screening test.

The staff concludes that, in general, 50 Hz is adequate to be used as the cut-off frequency for rugged equipment in screening process if the ZPA of the HRHF ISRS approaches 50 Hz. The staff considers RAI-SRP3.10-EMB-02 to be closed.

In RAI-SRP3.10-EMB-03, the staff requested that the applicant provide justifications for not performing additional low level testing (5 OBEs) for equipment identified as potentially sensitive to high frequency motion is located in an area with potential for high frequency seismic input motions. OBE testing requirements of Institute of Electrical and Electronics Engineers (IEEE) Std. 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," and NUREG-0800 Section 3.10 must be satisfied. The NRC's policy and staff's technical positions related to OBE issues are clearly delineated in SECY-93-087. The detailed response to RAI-SRP3.10-EMB-03 was initially submitted in a letter dated May 28, 2008, and later, a revised response was provided under the applicant's letter dated August 21, 2008. In the May 28, 2008, response, the applicant stated the HRHF screening test is not considered to be a qualification test. The HRHF screening test is intended as a supplemental test to the required seismic qualification performed in accordance with IEEE 344. As a result of further discussion with the staff, the applicant submitted its revised response on August 21, 2008. The applicant stated its HRHF screening test will be in compliance with the seismic test input requirements in IEEE Std. 344-1987 and Interim Staff Guidance defined in DC/COL-ISG-1. The five OBE (one-half SSE) and a minimum of one SSE AP1000 CSD ISRS test runs preceding the HRHF screening test are performed in compliance with IEEE Std. 344-1987. All of these test runs can be used to address seismic aging (fatigue) of the safety-related equipment in the high frequency exceedance region. Each test run will produce a number of peak stress magnitudes, which will have fatigue damage potential. OBE testing in the HF exceedance region was not significant because the cyclic fatigue of equipment (ten peak stress cycles per event) for equipment is more damaging in the frequency range below the HF exceedance region. The acceleration response in the HF exceedance region will produce very small displacements and lower number of high-stress cycles resulting in the overall equipment accumulative fatigue being less than or equal to that experienced during qualification testing.

The applicant's response to this RAI was partially acceptable. The applicant did not demonstrate OBE testing requirements of IEEE Std. 344-1987 and NUREG-0800 Section 3.10 (including SECY-93-087) were satisfied. Therefore, the staff followed up with RAI-SRP3.10-EMB-10 to continue resolution of the staff concerns. The staff's evaluation of the applicant's response to RAI-SRP3.10-EMB-10 is described later in this report. The staff considered RAI-SRP3.10-EMB-03 to be closed.

In RAI-SRP3.10-EMB/EEB-04, the staff requested that the applicant confirm battery chargers and inverters with digital components are included in the high frequency seismic screening process. The detailed response to this RAI was initially submitted in a letter dated May 28, 2008, and later, a revised response was provided under the applicant's letter dated August 21, 2008. The applicant stated electronic components such as those found in battery chargers, inverters, and solid state and microprocessor-based components are listed in Table 6.4.5-1, "Potential Sensitive Equipment List," of TR-115. The applicant further stated Table 3.11-1 of AP1000 DCD Revision 16 was reviewed to verify all potential high frequency (HF) sensitive AP1000 safety-related equipment were included in APP-GW-GLN-144 (TR-144) Table A-1, "Potential High Frequency Sensitive AP1000 Safety-Related Equipment." As a result of its review, the applicant identified additional equipment may be potentially HF-sensitive. Table 3I.6-2 of AP1000 DCD Revision 17 and Table A-1 of TR-144 have been updated to include the following additional equipment types: batteries, neutron detectors, radiation monitors and hot leg sample isolation limit switches. The remaining AP1000 safety-related equipment not high frequency sensitive is defined in APP-GW-GLN-144 Table A-2, "List of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive." Table 3I.6-3 of AP1000 DCD Revision 17 and Table A-3 of TR-144 include justifications for classifying the equipment as not HF-sensitive.

The staff has verified those electronic components in question are included in the tables mentioned above. The staff considers RAI-SRP3.10-EMB/EEB-04 closed.

In RAI-SRP3.10-EMB-05, the staff requested that the applicant provide justification for the conclusions addressing the use of existing test data in Section 6.4.7 (Summary and Conclusions) of TR-115. The detailed response to this RAI was initially submitted in a letter dated May 28, 2008, and later a revised response was provided under the applicant's letter dated August 21, 2008. The applicant stated the conclusions reached were based on the information presented in TR-115, Section 6.4.4 (Review of Existing Seismic Test Data). The test data in TR-115 represents existing the applicant seismic test data reviewed as part of the study to confirm seismic qualification to the AP1000 certified design ISRS envelops the HRHF seismic inputs for most applications. The applicant further stated that power spectral density (PSD) and other acceptable evaluation methods as defined in IEEE Std. 344-1987 are ways of determining energy content within a seismic test run. When available, PSD plots were used to evaluate seismic test data reported in Section 6.4.4 of TR-115. For the test data reported, energy content in the 25 Hz to 50 Hz frequency range was demonstrated by meeting at least one of the following criteria:

1. Test reports stated the seismic time history inputs were developed with content in the frequency range up to 50 Hz as a minimum.

2. The test response spectra (TRS) were shown to be amplified in the 25 Hz to 50 Hz frequency and were not caused by impact or test unit rattling.

3. PSD plots indicate energy content in the high frequency region.

Figures 1 through 6 of the applicant's response provide examples of test data that demonstrate frequency content in the 25 Hz to 50 Hz range.

The staff has examined Figures 1 through 6 and concluded that, for the existing test data reported, energy content in the 25 Hz to 50 Hz frequency range was demonstrated by meeting at least one of the criteria described above. Therefore, the staff considers RAI-SRP3.10-EMB-05 to be closed.

In RAI-SRP3.10-EMB-06, the staff requested that the applicant provide detailed evaluation comparisons for the reactor vessel internals response to the HRHF and CSDRS seismic input motions, and, also, the seismic anchor motion effects of the high frequency input motion. The detailed response to this RAI was submitted in a letter dated June 6, 2008. The applicant provided a comparison between the CSDRS results and HRHF results for various support (interface) loads within the reactor internals system model. The comparison indicates these support loads are reduced for HRHF evaluation when compared to the CSDRS analysis. The comparison also indicates CSDRS would control the cyclic loading demand. The applicant further stated the seismic anchor motion effects are included in the high frequency input motion study and, therefore, included in the evaluation.

The staff finds the applicant's response to be adequate in resolving its concerns relating to the comparison of the pertinent stress analysis results for the reactor internals system under the CSDRS and HRHF seismic input excitations. The applicant has also included the cyclic loading and seismic anchor motion effects in the HRHF evaluations. The staff considers RAI-SRP3.10-EMB-06 to be closed.

In RAI-SRP3.10-EMB-07, the staff requested that the applicant provide justification for concluding the reactor internals are representative of the primary mechanical components such that all others can be screened out, and also provided quantitative evaluation result for mechanical component other than reactor vessel internals to substantiate the justification. The detailed response to this RAI was submitted in a letter dated June 6, 2008. The applicant stated the mechanical components listed in Table 3.2-3 of the AP1000 DCD, Tier 2 must be designed for the SSE are those classified as Seismic Category, I and II. Among those equipment and components, the applicant stated many mechanical components and equipment that are safety-related are not high frequency sensitive as is some electrical equipment. Therefore, it is only necessary to evaluate a representative sample of mechanical components and equipment. The applicant stated the reactor vessel is representative of a mechanical component with complex internals that was evaluated as part of the HRHF evaluation. The seismic response of this component is considered representative of other mechanical components. The reactor internals were chosen for evaluation because this is an important component related to safety, and the reactor internals are representative of other component internals. It is, therefore, not necessary to perform further analysis of other mechanical components and equipment for the HRHF earthquake excitations.

The staff concludes reactor internals are relatively complex and contain broader natural frequencies than other mechanical components. The staff considers the applicant's response adequate in justifying that reactor internals can be considered as representative of ASME safety-related mechanical components and the equipment for high frequency evaluation. Therefore, the staff considers RAI-SRP3.10-EMB-07 to be closed.

In RAI-SRP3.10-EMB-08, the staff requested that the applicant justify the use of the required input motion (RIM) curve of IEEE Std. 382-1996, "IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies With Safety-Related Functions for Nuclear Power Plants," for qualification of line-mounted equipment (e.g., valves) for HRHF response spectra with exceedance, or to provide methodologies would be acceptable for the case of HRHF excitation. The detailed response to this RAI was initially submitted in a letter dated May 28, 2008, and later Revision 1 response was provided in a letter dated August 21, 2008. Revision 2 of the applicant's response was submitted in a letter dated May 27, 2009. The applicant stated it was performing seismic qualification of safety-related SSCs based on AP1000 CSDRS. The HRHF screening is a functional verification test in compliance with Interim Staff Guidance defined in COL/DC-ISG-1 to verify potential high frequency sensitive safety-related equipment will perform its function as required under HRHF seismic demand response spectra. The HF screening is a supplemental evaluation to the required seismic qualification methods performed in accordance with IEEE Std. 344-1987 for those plants that have potentially high frequency sensitive equipment and components with high frequency exceedance of their CSDRS.

The applicant stated in those instances where the seismic qualification of line-mounted equipment (e.g., valves and their appurtenances) are potential HRHF-sensitive components, seismic testing performed in compliance with Figure 6 (RIM curve) of IEEE Std. 382-1996 will be extended out for one additional octave to 64 Hz.

AP1000 DCD Tier 2, Section 3.7.3.5.1 defines rigid components such as rigid valves as the following: "A rigid component (fundamental frequency >33 hertz), whose support can be represented by a flexible spring, can be modeled as a single degree of freedom model in the direction of excitation (horizontal or vertical directions)." When dealing with HRHF sites we should refrain from using the wording rigid equipment or rigid components because it can differ between the AP1000 CSRDS and HRHF sites. Seismic qualification of safety-related equipment by analysis will be addressed over the range of interest up to the cutoff frequency of the AP1000 certified design ISRS. In most instances a dynamic analysis or a static coefficient analysis using the peak of the applicable response spectra at the mounting location of the equipment will be used.

The applicant further noted in its Revision 2 Response, dated May 27, 2009, AP1000 DCD Tier 2, Appendix 3I, Table 3I.6-3 includes a list of AP1000 safety-related equipment and mechanical equipment not high frequency sensitive. Notes 1 and 2 of the table identify the requirement for performing seismic RIM testing of line-mounted equipment out to 64 Hz.

Based on the review of the applicant's documents, DPC/NRC2144, DCP/NRC2235, and DCP/NRC2503, the staff determined the applicant has adequately addressed the questions raised in this RAI. The staff has also verified the conclusion of the applicant's response to this RAI has been documented in Notes 1 and 2 of Table 3I.6-3 in Tier 2 document Appendix 3I of AP1000 DCD Revision 17. Therefore, the staff considers RAI-SRP3.10-EMB-08 to be closed.

In RAI-SRP3.10-EMB-09, staff requested that the applicant discuss the basis for deleting references to dampers in Section 3.10. In several locations in Section 3.10 of AP1000 DCD and Revision 17, the applicant has replaced the reference to safety-related dampers with a reference to safety-related valves; Section 3.10.2.2 is an example. The applicant's response to this RAI was submitted in a letter dated May 28, 2008. The applicant stated for the AP1000 design, there are no safety-related dampers. The term "dampers" was used in error. Changes were made in Section 3.10 of AP1000 DCD and Revision 17 to correctly identify the subject equipment as safety-related valves. The staff considers the applicant's response to be acceptable. Therefore, RAI-SRP3.10-EMB-09 is closed.

In the revised response to NRC RAI-SRP3.10-EMB-03 dated August 21, 2008 (ADAMS Accession Number ML082390116), the applicant indicated the five OBE (one-half SSE) and a minimum of one SSE AP1000 CSD ISRS test runs preceding the HRHF screening test were performed in compliance with IEEE Std. 344-1987. The staff understands the same specimen is used for all these test runs. The applicant also indicated all of the CSDRS test runs can be used to address seismic aging of the equipment in the high frequency exceedance region.

In RAI-SRP3.10-EMB-10, as a follow-up to the August 2008 response to RAI-SRP3.10-EMB-03, the staff requested that the applicant provide justifications including the results from calculations that show seismic qualification of electrical/electronic equipment by tests for AP1000 CSDRS design spectra can be considered as equivalent to or more than 5 OBE peak stress cycles for HRHF spectra. This should be done using bounding AP1000 ISRS generated from CSDRS and bounding ISRS generated from HRHF Spectra and following the guidelines as delineated in Annex D of IEEE 344-1987. The staff also requested that the applicant document the conclusion of the comparison result of CSD ISRS and HRHF ISRS peak stress cycles in DCD Section 3I.6.4. In its response dated October 17, 2008 and March 5, 2009, the applicant stated that the AP1000 safety-related equipment will be seismically- qualified to the AP1000 CSD ISRS associated with the mounting location of the equipment as a minimum. Seismic qualification testing will consist of five AP1000 ISRS OBEs followed by one SSE as a minimum. The OBE level will be at least one-half the SSE level. The OBE testing is used to account for vibration aging and address low-cycle fatigue of equipment prior to SSE testing. The applicant stated cyclic fatiguing of equipment for the HRHF exceedance area can be adequately addressed by performing five AP1000 ISRS OBE (one-half the SSE) and a minimum of one

SSE seismic test runs in compliance with IEEE Std. 344-1987 prior to performing the supplemental HRHF screening test.

The applicant has performed an evaluation to demonstrate OBE testing in the high frequency exceedance range is adequately addressed by AP1000 CSD ISRS seismic qualification testing (5 OBE and 1 SSE). The evaluation compared the peak stress cycles resulting from five one-half SSE events from AP1000 HRHF ISRS to the peak stress cycles resulting from five one-half SSE events and one full SSE event from AP1000 CSD ISRS using the guidelines defined in Annex D of IEEE Std. 344-1987. The applicant's evaluation of AP1000 CSD ISRS peak stress cycles to the AP1000 HRHF ISRS peak stress cycles is documented in Westinghouse Calculation CN-EQT-08-35/APP-GW-S2C-002. The evaluation of AP1000 CSD ISRS peak stress cycles to the AP1000 HRHF ISRS peak stress cycles was performed for two AP1000 plant elevations; the AP1000 NI Auxiliary and Shield Building (ASB) at or below 135 feet elevation and the AP1000 CIS at or below 40.9194 m (134.25 ft) El.

The peak stress cycles in each direction were determined based on the ZPA of the 1/2 SSE HRHF ISRS and the 1/2 SSE and SSE CSD ISRS acceleration time histories normalized to the same ZPA value to demonstrate equivalency of results. Results of the cycle counting in compliance with guidelines defined in Annex D of IEEE Std. 344-1987 are summarized in Table 1 of the applicant's letter.

The applicant concluded that the completed evaluation has demonstrated the peak stress cycles resulting from five one-half SSE events using the AP1000 HRHF ISRS are equivalent to or enveloped by the peak stress cycles resulting from five one-half SSE events and one full SSE event using the AP1000 CSD ISRS.

The staff has reviewed the applicant's responses as stated above. The staff concludes the applicant has adequately demonstrated by calculations that the peak stress cycles resulting from five one-half SSE events using the AP1000 HRHF ISRS are equivalent to or enveloped by the peak stress cycles resulting from five one-half SSE events and one full SSE event using the AP1000 CSD ISRS. The applicant's response also included the proposed revision to the DCD. The staff finds the proposed revision acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In RAI-SRP3.10-EMB-11, the staff pointed out that the applicant's evaluations from the envelope response spectra (ERS) for the Central and Eastern U. S. rock sites (HRHF foundation response spectra exceeds CSDRS for frequencies above about 15 Hz) presented in TR-115, Revision 2, indicated some resulting FRS are higher than those presented in Revision 0 and Revision 1. Some resulting HRHF FRS exceeds the CSDRS FRS significantly in the frequency range from 6 Hz to 60 Hz. The applicant is requested to demonstrate all AP1000 safety-related mechanical and electrical equipment (not limited to HF-sensitive equipment only) are seismically qualified to the RRS at the equipment/device locations to meet the requirements of GDC 2 as a result of TR-115, Revision 2.

The detailed response to this RAI was initially submitted in a letter dated August 6, 2010, and later, Revision 1 response was provided under the applicant's letter dated August 23, 2010. The applicant stated the equipment seismic qualification information provided in Revision 2 of TR-115 was developed using the latest revision of the CSDRS and HRHF. The equipment will be qualified to the latest AP1000 CSDRS applicable to its mounting location. Potential HRHF sensitive equipment will be subjected to a high frequency screening test consistent with the guidance in ISG-1 after completion of CSDRS testing. Appendix 3I of the AP1000 DCD provides two tables that separate the AP1000 safety related electrical and mechanical equipment into two categories: (1) Table 3I.6-2 includes equipment that is potentially high frequency sensitive and (2) Table 3I.6-3 contains equipment not high frequency sensitive. The applicant stated seismic qualification for both categories is performed using the current revision of the CSDRS ISRS associated with the mounting locations of equipment based on the guidance of IEEE Std. 344-1987. After completing CSDRS ISRS seismic qualification testing, the potential high frequency sensitive equipment will be subjected to a HRHF screening test to the HRHF ISRS associated with the mounting location of the equipment as a minimum. To demonstrate acceptability for both CSDRS and HRHF testing, the test response spectra must envelop the CSDRS ISRS with margin over the frequency range of interest in compliance with IEEE Std. 344-1987. If the HRHF screening test cannot demonstrate the equipment to be acceptable, then the safety related equipment will be removed or modified and additional testing or justification will be required.

The applicant noted that, at locations where HRHF response spectra show exceedance of the CSDRS for category (2) equipment (not high frequency sensitive) in Table 3I.6-3 of AP1000 DCD, further evaluations would be performed to verify the existing qualification is adequate. The applicant further stated that, in the event that the CSDRS and/or HRHF response spectra would be revised after the qualification program had been completed, a reconciliation effort were performed to verify the CSDRS and HRHF testing was still valid. The reconciliation effort may result in re-qualification activities and qualification documentation revisions.

The seismic qualification testing for both CSDRS and HRHF spectra for AP1000 safety-related equipment will be documented in equipment qualification document packages as described in Appendix 3D of the AP1000 DCD. These qualification document packages will be used to satisfy the equipment's seismic Category I ITAAC described in Tier 1 of the AP1000 DCD. The applicant's response to RAI-SRP3.10-EMB-11 also included the proposed revision to the DCD (Section 3I.6.4, "Electrical and Electro-mechanical Equipment").

Based on the aforementioned response, the staff finds the applicant's response to RAI-SRP3.10-EMB-11 and the proposed revision to AP1000 DCD is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.10.2 Conclusion

The staff reviewed the proposed changes related to Section 3.10 of AP1000 DCD, including Appendix 3I to the DCD. On the basis that the AP1000 mechanical and electrical equipment continue to meet all applicable acceptance criteria and procedures for seismic qualification of mechanical electrical equipment in accordance with the guidance of NUREG-0800 Section 3.10, RG 1.100, SECY-93-087, and DC/COL-ISG-1, the staff finds the changes to Section 3.10 of AP1000 DCD are acceptable. The staff finds the AP1000 design provides adequate assurance that AP1000 seismic Category I equipment will function properly under the effects of earthquake motions, and the acceptance criteria for the AP1000 design meet the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, and 30; and 10 CFR Part 50, Appendix S.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

In Revision 17 to the AP1000 DCD Tier 2, the applicant modified Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The objective of environmental qualification (EQ) is to reduce the potential for common failure due to specified environmental conditions and seismic events, and to demonstrate the equipment within the scope of the EQ Program is capable of performing its intended design safety function under all conditions including environmental stresses resulting from design bases events.

3.11.1 Evaluation

In Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," of NUREG-1793, the staff described its review of the description of the EQ Program for the AP1000 design. The regulatory basis for the NRC review of the design certification information is documented in NUREG-1793. The regulatory basis for the proposed changes to the AP1000 DCD is the same as specified in NUREG-1793.

In NUREG-1793, the staff concluded the program described for environmentally qualifying electrical equipment important to safety and safety-related mechanical equipment in support of the AP1000 DC complied with the requirements for 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," and other relevant requirements and criteria.

Since the issuance of NUREG-1793, the NRC has determined a COL applicant referencing the AP1000 design needs to fully describe EQ and other operational programs as defined in SECY-05-0197. RG 1.206 provides guidance for a COL applicant in preparing and submitting its COL application in accordance with the NRC regulations. For example, Section C.IV.4 in RG 1.206 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs that need to be included in the FSAR in a COL application to support a reasonable assurance finding of acceptability. A COL applicant may rely on information in the applicable DC to help provide a full description of the operational programs for the COL application. At a public meeting on March 26 and 27, 2008, the applicant indicated its intent to revise the AP1000 DCD to resolve issues common to COL applicants implementing the AP1000 design. Therefore, the staff reviewed Revision 17 to the AP1000 DCD, Section 3.11, including DCD changes intended to minimize the supplemental information necessary to be provided by COL applicants in fully describing their operational programs in support of the COL applications. As described below for specific review areas, the staff finds the revision to the AP1000 DCD continues to provide an acceptable description of the EQ Program sufficient for the AP1000 Design Certification in accordance with the NRC regulations.

A COL applicant referencing the AP1000 design will be responsible for fully describing the EQ operational program in support of its COL application. A COL applicant may reference the provisions in the AP1000 DCD as part of its responsibility to fully describe the EQ operational program. The staff will evaluate the full description of the operational EQ Operational Program provided by a COL applicant during review of the COL application consistent with RG 1.206 and NUREG-0800 Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment."

Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," in AP1000 DCD Tier 2 presents information to demonstrate the mechanical and electrical portions of plant safety systems are capable of performing their designated functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. AP1000 DCD Tier 2, Appendix 3D, "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment," describes the methodology to be used to qualify equipment for nuclear power plants with the AP1000 reactor design. During the March 26 and 27, 2008, public meeting, the applicant stated procurement specifications were being prepared for safety-related equipment to be used in the AP1000 reactor design. In RAI-SRP3.11-CIB1-01, the staff requested that the applicant describe the implementation of the methodology for environmental qualification of safety-related mechanical equipment to be used in the AP1000. In its response to this RAI, in a letter dated May 30, 2008, the applicant described its EQ Program for safety-related mechanical equipment. The applicant stated safety-related functions of mechanical equipment shall be shown to be acceptable under the required operating conditions and environmental parameters. Further, the AP1000 harsh and mild environmental conditions will be supplied to the vendor in the design and qualification specifications.

On October 14 and 15, 2008, the staff conducted an audit of design and procurement specifications, including environmental qualification, for pumps, valves, and dynamic restraints to be used for the AP1000 reactor at the applicant office in Monroeville, Pennsylvania. The staff found the applicant had included ASME Standard QME-1-2007 in its design and procurement specifications for AP1000 components, including ASME QME-1-2007, Appendix QR-B, “Guide for Qualification of Nonmetallic Parts.” Further, AP1000 DCD Tier 2 (Revision 17), Section 5.4.8.3, “Design Evaluation,” states the requirements for qualification testing of power-operated active valves are based on ASME Standard QME-1-2007 as listed in AP1000 Tier 2, Section 5.4.16, “References.” In a memorandum dated November 6, 2008, the staff documented the results of the audit with specific open items. In a letter dated January 26, 2010, the applicant discussed its plans to address the October 2008 audit findings. Resolution of the audit findings were tracked as Open Item OI-SRP3.11-CIB1-01. In a letter dated February 23, 2010, the applicant provided its response to Open Item OI-SRP3.11-CIB1-01. In particular, the applicant stated the valve design specifications indicate active valves will be qualified in accordance with ASME Standard QME-1-2007. The applicant also provided planned changes to the AP1000 DCD to ensure consistency in the EQ provisions and tables.

As discussed in Section 3.9.6 of this SER, the staff conducted a follow-up audit at the applicant’s office in Rockville, Maryland, on May 17, 2010, to review the revisions to the design and procurement specifications prepared since the October 2008 audit. Based on the May 2010 audit, the staff found the applicant has updated the design and procurement specifications to address NRC comments provided during the October 2008 audit. For example, the staff found the design and procurement specifications require the application of ASME Standard QME-1-2007 for the qualification of mechanical equipment to be used in an AP1000 reactor. Further, the staff found the equipment qualification specification mandates non-metallic components be environmentally qualified using the provisions of Appendix QR-B to ASME Standard QME-1-2007. Based on the follow-up audit, Open Item OI-SRP3.11-CIB1-01 is closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 3.11.5, “Combined License Information Item for Equipment Qualification File,” in Revision 17 to the AP1000 DCD states the COL holder will define the process and procedures for which the equipment qualification files will be accepted from the applicant and how the files will be retained and maintained in an auditable format for the period that the equipment is installed and/or stored for future use in the nuclear power plant. In RAI-SRP3.11-CIB1-02, the staff requested that the applicant specify the necessary actions for the COL applicant to establish the process and procedures for accepting, maintaining, and storing equipment qualification files. In its response to this RAI in a letter dated May 30, 2008, the applicant stated it will act as the agent for the COL holder during the equipment design phase, equipment selection and procurement phases, equipment qualification phase, plant construction phase, and ITAAC inspection phases. The applicant indicated the COL applicant will provide supplemental information to fully describe the process for retention and maintenance of the EQ documentation for the operational life of the plant. The staff considers the RAI response clarifies the role of the applicant in the EQ process; RAI-SRP3.11-CIB1-02 is closed.

The staff reviewed the revisions to the AP1000 DCD with respect to the EQ Program for electrical equipment important to safety and safety-related mechanical equipment. The staff finds the changes are generic and are expected to be applicable to all COL applications referencing the AP1000 certified design.

3.11.2 Conclusion

The staff concludes that the proposed changes to the AP1000 DCD continue to satisfy the NRC regulations for electrical and mechanical equipment within the scope of the EQ Program for the AP1000 design. The revision to the AP1000 DCD provides sufficient information to satisfy 10 CFR Part 50 and 10 CFR Part 52 for the EQ Operational Program for electrical and mechanical equipment to be used at an AP1000 nuclear power plant and are, therefore, acceptable. Further, the staff concludes the AP1000 DCD changes related to the EQ Operational Program are generic and are expected to apply to all COL applications referencing the AP1000 DC.

3.12 Piping Design

The AP1000 DCD, Revision 15 was approved by the staff in the certified design. In the AP1000 DCD Revision 17, the applicant proposed the completion of Design Specification and Reports which is COL Information Item 3.9.2 in Sections 3.9.3.1.4, 3.9.3.1.5, 3.9.3.4 and 3.9.8.2 and 3.9.8.6 of the DCD. In addition, COL Information Item 3.9-6 would be closed in Section 3.9.8.6.

In Appendix 3I, the applicant proposed to address HR sites that show higher amplitude at high frequency than the CSDRS. In Appendix 3C, the applicant proposed to remove the containment interior building structure and the surge line piping from the original reactor coolant loop (RCL) model and provided more accurate description for the RCL model and analysis methods.

In Section 3.9.3.1.2, the applicant revised piping lines connected to the RCS from not susceptible to thermal stratification, cycling or striping (TASCS) to susceptible to TASCS. The applicant added clarification to Sections 3.9.3.1.4 and 3.9.3.1.5. The applicant proposed changes to the requirement for the welded connections of ASTM A500 Grade B tube steel members as described in Section 3.9.3.4. In Section 3.9.3.4, the pipe support deflection limit and pipe support stiffness values used in the piping analysis were clarified. Clarification was added in Section 3.9.8.6 to address COL information item related to piping benchmark program. Lastly, the applicant proposed changes in Section 3.9.8.2 to remove piping DAC from the DCD.

3.12.1 Evaluation

The staff reviewed the proposed changes to the piping design in the AP1000 Revision 17 in accordance with the guidance in the NUREG-0800 Section 3.12, "ASME Code Class 1, 2, and 3 Piping System, Piping Components and their Associated Supports." The regulatory basis for Section 3.12 of the AP1000 DCD is documented in NUREG-1793.

3.12.1.1 Design Specification and Reports

In Section 3.9.8.2 the applicant stated "COL holder referencing the AP1000 design will have available for NRC audit the design specifications and as-designed reports prepared for major ASME Section III components and ASME Code, Section III piping." The statement implied the COL applicant may not complete the piping design prior to issuance of a COL.

On February 8, 2008, the applicant issued a letter related to schedule for piping design document review. In this letter, the applicant stated "It is the intention of Westinghouse that design documents related to DAC and COL Information Item will be available for NRC review during the period scheduled for the NRC review of the design certification amendment. It is expected information will be available for NRC review to permit the resolution, closure, or removal of the DAC and COL Information Item."

In RAI-SRP3.12-EMB-4, the staff questioned whether the applicant would complete the as-designed piping analyses and design reports by December 2008 as stated in the February 8, 2008, letter. The staff requested that the applicant revise the DCD to reflect the design completion or propose a method and schedule to resolve the piping DAC issue and update the DCD to reflect the proposed alternative.

By letter dated June 20, 2008, the applicant responded that it intended to have the design documents for the risk-significant piping packages identified by the NRC available during the review of the design certification amendment. The DCD would be revised to reflect the expected completion of the piping design. It was also expected the NRC's review of these documents would permit the resolution, closure, or removal of the DAC and the COL Information Item. The staff reviewed the applicant's response and its follow-up letter, which indicated the schedule for the risk-significant piping packages would be completed by June 30, 2009, in order to resolve the piping DAC.

By letter dated April 1, 2010, the applicant stated that it would not be able to complete the piping analysis for the previously specified risk significant piping packages to support the DC amendment. In this letter, the applicant revised DCD Sections 3.9.8.2 and 3.9.8.7 to address all aspects related to as-designed piping design specification, design reports, and analysis. Additionally, the applicant revised Section 14.3.2.2 of DCD to address DAC/ITAAC closure process for OI-SRP3.12-EMB-04.

The applicant proposed the piping DAC become COL Information Item 3.9-7 in the DCD for the DC amendment. COL Information Item 3.9-7 would state the COL applicant needs to complete the as-designed piping analysis for the identified risk significant piping packages to close the piping DAC. The COL information item may be addressed by the COL applicant in a manner that complies with NRC guidance provided in RG 1.215, and outlined in Appendix 14.3A of the DCD. The applicant's use of piping DAC was previously approved and certified in Revision 15 of the DCD and documented in NUREG-1793. The staff finds this acceptable.

By letter dated August 23, 2010, the applicant provided a Revision 2 response to OI-SRP3.12-EMB-04 to address RAI-SRP3.12-EMB1-09 that is documented in Section 3.12.1.3 of this SER. In this letter, the applicant revised Sections 3.9.8.2 and 3.9.8.7 to further clarify as-designed design specifications, design reports, and analysis will be the responsibility of the COL applicants. The applicant also clarified the availability of the piping design information and design reports will be identified to the NRC. The staff has determined these clarifications for the COL applicants' activities are acceptable. In that letter, the applicant added a Table 3.9-20, which describes piping packages chosen to demonstrate piping design for piping DAC closure (in addition to Class 1 lines larger than (2.54 cm (1 in) in diameter). The staff reviewed this table and concluded these piping packages and Class 1 lines do represent the Class 1, 2, and 3 piping packages and would be able to demonstrate piping design for piping DAC closure. Therefore, the staff finds this acceptable.

In this letter, the applicant also provided a markup of Section 14.3A of the DCD for DAC/ITAAC closure process as follows:

> 14.3A Design Acceptance Criteria / ITAAC Closure Process
>
> DAC (Design Acceptance Criteria) are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a DC. DAC are to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as a part of the ITAAC performed to demonstrate the as-built facility conforms to the certified design. (SECY-92-053, "Use of Design Acceptance Criteria during 10 CFR Part 52 Design Certification Reviews")
>
> There are three process options for DAC/ITAAC resolution:
>
> - Resolve through amendment to design certification
> - Resolve as part of COL review
> - Resolve after COL is issued
>
> In the first two options, the applicant will submit the design information and the NRC will document its review in a safety evaluation. In the third option, the COL holder notifies the NRC of availability of design information and the staff will document its review in an inspection report.
>
> Should the third option be implemented for the first standard AP1000 plant, subsequent COL applicants may reference the first standard plant closure documentation and close the DAC/ITAAC under the concept of "one issue, one review, one position," identified in NRC guidance.
>
> Additionally, the applicant may submit licensing topical reports for NRC review of the material supporting the DAC/ITAAC closure and request the NRC to issue a safety evaluation in conjunction with a closure letter or inspection report concluding the acceptance criteria of the DAC/ITAAC have been met. Subsequent COL applicants may reference these reports and NRC closure documents in effort to close the DAC/ITAAC.
>
> For technical areas where DAC/ITAAC applies in the DC rule, COL applicants will provide an ITAAC and associated closure schedule indicating the approach to be applied.
>
> For subsequent COL applicants following the first standard AP1000 plant, the application could reference the existing DAC/ITAAC closure documentation for the first standard plant.

NRC guidance for DAC/ITAAC is provided in RG 1.206, Section C.III.5. Further information on the staff's position of DAC/ITAAC being used as part of the 10 CFR Part 52 review process is provided in SECY-92-053.

The staff reviewed and evaluated the applicant's letter addressing OI-SRP3.12-EMB-4. On the basis that the applicant-proposed position provides no change from the approved design (the use of Piping DAC had been approved in DCD Revision 15) and better defines a plan for closure later in the construction period, the staff finds this acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In RAI-SRP3.12-EMB-5, the staff questioned how the COL holder would complete verification of the thermal cycling and stratification loading considered in the stress analysis as discussed in Section 3.9.3.1.2 prior to fuel load. The applicant had not provided a specific monitoring program for verification of thermal cycling and stratification loading condition of the automatic depressurization Stage 4 lines and the passive residual heat removal line. These two lines are susceptible to thermal stratification as described in Section 3.9.3.1.2 of DCD. If verification could not be completed prior to fuel load, the applicant was requested to provide alternatives.

In a letter dated June 20, 2008, the applicant responded that Section 3.9.8.2 deals with design specifications and design reports and the requirement to perform a reconciliation/analysis for the as-built piping. The intent of the phrase in parenthesis (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in Section 3.9.3.1.2) was to verify "dimensional/layout/support differences" identified in an as-built walk-down were considered in thermal cycling/stratification as well as the standard portion of the piping analysis. The monitoring program identified in Section 3.9.8.5 was a one-time requirement for the surge line and was not related or applicable to Section 3.9.8.2. Thermal cycling and stratification loading were to be evaluated by analysis and if the as-built dimensions, layout, or supports on the piping lines changed as the result of construction, a reconciliation of the stratification analysis was to be performed. The staff reviewed the clarification provided in the response and determined it was acceptable.

3.12.1.2 Closure of COL Information Item 3.9-6 (Piping Benchmark Program)

The original COL information item commitment stated the COL applicant will implement a benchmark program as described in Section 3.9.1.2 if a piping computer program other than the one used for design certification is used. The piping benchmark problems identified in Reference 20 for the Westinghouse AP600 are also representative of AP1000 and can be used for the AP1000 piping benchmark program if required.

In Section 3.9.8.6, the applicant proposed to close out the COL Information Item 3.9-6. The applicant stated the combined license information requested in this subsection had been completely addressed in TR-15, "Benchmark Program for Piping Analysis Computer Programs," APP-GW-GLR-006, March 2006, and no additional work was required by the combined license applicant to address the combined license information requested in this subsection.

The staff reviewed TR-06, which stated all piping analysis performed for AP1000 was being completed using only programs that had already been benchmarked to NRC's satisfaction. PIPESTRESS, GAPPIPE, WECAN and ANSYS require no additional benchmarking by the COL Applicant. On the basis that the above mentioned computer codes have been accepted by the staff and other analysis codes are not being used for piping analysis, the staff finds this change acceptable and COL Information Item 3.9.6 is closed.

3.12.1.3 Evaluation for High Frequency Seismic Input

The staff reviewed the applicant's Appendix 3I of DCD Revision 17, and its supporting document, TR-115. However, the seismic input has been identified in Section 3.7.3 as inadequate due to mathematical model error. The applicant revised TR-115 with adequate seismic input. The staff reviewed the revised document, TR-115, Revision 2.

TR-115, Revision 2 states the HRHF exceeds the CSDRS for frequencies above about 15 HZ and the representative piping considered is based on high frequency participation as indicated in Section 6.3.1 of TR-115.

The staff noted the floor response spectra exceedances were in the region for frequencies around 6-12 Hz as shown in Figure 6.3.2.2-1 through 6.3.2.2-3 of TR-115. The staff identified the selected representative piping systems only addressed high frequency piping systems. TR-115 failed to address all other piping packages related to the FRS exceedances for frequencies around 6-12 Hz. The staff reviewed sample piping system selection criteria and determined the selection criteria was not adequate. The staff also noted TR-115, Revision 2 did not address support load increase due to the response spectra exceedance. The staff issued OI-SRP3.12-EMB1-09 to ask the applicant to address all piping packages for the FRS exceedances around 6-12 Hz and piping support load increases.

In a letter dated August 17, 2010, the applicant provided the following response:

> HRHF GMRS effects on ASME Class 1, 2 and 3 piping systems will be evaluated as part of the Piping DAC, captured as COL Information Item 3.9-7 in the DCD (See response to Open Item OI-SRP3.12-EMB-4 R1 in Letter No. DCP_NRC_002845).
>
> Areas of exceedance of the CSDRS will be addressed for the entire frequency range.
>
> The following will be considered in the evaluation:
> - Piping Qualification
> - Support Loads
> - Valve Accelerations
> - Valve End Stresses
> - Equipment Nozzle Loads
>
> Impacts on the following evaluations will be considered:
> - Pipe Rupture Hazards Analysis
> - LBB
> - Piping and Component Fatigue Analysis"

The applicant also provided a markup of revised Table 3.9-19 and revised Appendix 3I to state ASME Class 1, 2 and 3 piping systems will be evaluated for the HRHF GMRS and this evaluation is within the scope of the piping DAC.

The staff reviewed the applicant's response and identified COL Information Item 3.9-7 is not listed in the Table 1.8-2 of DCD. The staff requested that the applicant address COL Information Item 3.9-7 (as-designed piping analysis). In a letter dated August 23, 2010, the applicant provided a markup of revised Table 1.8-2 by adding this COL Information Item to address staff's concern. In this letter, the applicant also revised its DAC/ITAAC closure process. The evaluation of DAC/ITAAC closure process is documented in Section 3.12.1.1.

In the markup of revised Appendix 3I of DCD, the applicant stated the COL will evaluate the HRHF GMRS effects of the piping systems by changing the piping system screening criteria from at least 2 piping analysis package to all ASME Code Class 1, 2, and 3 piping systems. The staff determined the revised position meets design requirements of GDC 2. Therefore, the staff finds this acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.12.1.4 Reactor Coolant Loop Analysis Methods

The staff reviewed the proposed change in Appendix 3C of DCD Revision 17, and its supporting document TR-13, "Safety Class Piping Design Specifications and Design Reports Summary," APP-GW-GLR-013, Revision 1, May 2007. The proposed change would remove the containment interior building structure and the surge line piping from the reactor coolant loop model description. The staff also reviewed the applicant's proposed change related to time history analysis in Section 3.9.3.1.4 of DCD Revision 17. The applicant clarified that unless appropriate time-history seismic input from the building is provided at multiple supported locations, the containment internals structure is included in the system-coupled model in the time-history analysis. The staff agreed that a containment interior building structure is not required because the seismic inputs to the RCL model are provided at all of the building attachments to the RCL.

TR-13 identifies that pressurizer surge-line piping is to be analyzed in APP-RCS-PLR-040 as listed in Table 2. In RAI-SRP3.12-EMB-6, the staff noted that the RCL analysis did not couple the branch lines such as the pressurizer surge line. Section 3.7.3.8.1 of the DCD states that "if the ratio of the run piping outside diameter to the branch piping outside diameter (nominal pipe side) exceeds or equals 3.0, the branch piping can be excluded from the analysis of the run piping." Several branch lines do not meet this ratio, and therefore, should be included in the RCL piping analysis. The staff requested that the applicant explain this discrepancy and take action to address this DCD conformance issue.

In a letter dated December 23, 2008, the applicant responded as follows:

> The branch piping has been excluded from the reactor coolant loop analysis because the criteria in Subsection 3.7.3.8.1 do not apply to the hot and cold leg piping. Just as attached piping is excluded from primary equipment models, the branch piping of the surge line, automatic depressurization system Stage 4 (ADS4), RNS suction line, and several smaller lines are excluded from the analysis of the hot and cold leg piping.
>
> The reactor coolant loop (APP-RCS-PLA-050) is unique in that the stiffness and mass characteristics are closer to that of equipment than a typical piping analysis package. With the relatively short run length, comparatively large pipe diameter and pipe thickness, both the cold and hot legs have much less flexibility than a typical run length of pipe. The large interplay of the hot and cold leg piping with the reactor pressure vessel and steam generator extends the boundary of the piping analysis package to include primary equipment as well as primary loop piping.

> No non-conformance exists because the reactor coolant loop piping is treated as a rigid piece of equipment (fundamental frequency greater than 33 Hz) and not a flexible pipe.

In this letter, the applicant also submitted the DCD revision for Section 3.7.3.8.1 and Appendix 3C to reflect that the branch piping is excluded from the RCL analysis.

The staff reviewed the RCL layout configuration, which showed a total length of the 95.25 cm (37.5 in) outside diameter hot leg pipe to be approximately 6.09 m (20 ft), which should conform to a rigid body motion. On the basis that the applicant has performed a calculation to demonstrate that AP1000 RCL piping is rigid and has a fundamental frequency much higher than 33 Hz, the staff finds this acceptable.

On the basis of above discussion, the staff found that the proposed change is acceptable to reflect the RCL model used in the loop analysis.

3.12.1.5 Remove Piping Design Acceptance Criteria (DAC)

The staff reviewed the proposed changes in the introduction of DCD Revision 17 and related Tier 2 Section 3.9.8.2 and Table 3.9-19. The staff determined that risk-significant piping design packages would have to be completed in order to resolve or remove reference to piping DAC.

DCD Section 3.9.8.2 was revised to reflect the design completion by indicating that as-designed design specifications and design report for the major ASME Code, Section III components and piping are available for NRC review.

During the period October 20-24, 2008, the staff performed an on-site review, at the applicant's headquarters, of ASME Code Class 1 piping and support design with the intent to resolve piping DAC. During this review, the staff found that the applicant had not completed risk-significant ASME Class 1 piping analysis packages. On the basis that the risk-significant piping analyses had not been completed, the staff cannot remove piping DAC at this time. The onsite review summary is documented in a letter dated December 30, 2008.

In a letter dated January 19, 2009, the applicant submitted AP1000 piping DAC Analysis Schedule. In this letter, the applicant stated that the AP1000 ASME Code, Section III, Class 1, 2, and 3 piping analysis packages are rescheduled to be completed by June 30, 2009. The applicant will inform the staff when it is ready for another on-site review for Class 1, 2, and 3 risk-significant piping analysis to complete resolution of the piping DAC. This is Open Item OI-SRP3.12-EMB-4.

In a letter dated April 1, 2010, the applicant stated that it would not be able to complete the piping analysis for the previously specified risk significant piping packages to support the DC amendment. The resolution and evaluation are discussed and evaluated in Section 3.12.1.1 of this report.

3.12.1.6 Change Component and Piping Support Weld Connections Requirement

The staff reviewed the proposed changes in Tier 2 Section 3.9.3.4 of DCD Revision 17. Section 3.9.3.4 stated that the welded connections of ASTM 500 Grade B tube steel members satisfy the requirements of the AISC "Load and Resistance Factor Design (LRFD) Specification for Steel Hollow Structure Sections," dated November 10, 2000. NUREG-0800 Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," Acceptance Criteria 2, identified applicable steel structure codes, standard, and specifications. The applicant proposed LRFD Specification is not listed as acceptable. NUREG-0800 proposed "ANSI/AISC N690-1994, including Supplement 2 (2004)" as an acceptable specification. ANSI/AISC N690-1994 including Supplement 2 (2004) has been accepted by the NRC as ASME Code Case N-570-2. The later LRFD version of AISC N690, ASME Code Case N-721, has not been accepted by the NRC. The staff noted that the NRC's current acceptable specification is based on allowable stress design (ASD) specification. Further, the LRFD method has not been approved for use in the design of new reactor nuclear facilities. In RAI-SRP3.12-EMB-8, the staff requested that the applicant identify differences between the two methods and show equivalency with respect to NUREG-0800 acceptable specification or provide alternatives to satisfy the acceptance criteria.

In a letter dated July 30, 2009, the applicant stated that the AP1000 component and piping support designs satisfy the requirements of the ASME Code Section III, Subsection NF and the requirements in the DCD on the welding of members fabricated on tube steel are in addition to the requirements in Subsection NF. These requirements are not considered to be an alternative to the Subsection NF requirements. On the basis that the applicant meets the requirements of ASME Code Section III, Subsection NF, any additional requirements imposed by the applicant shall provide additional level of quality and safety. Therefore, the staff finds this acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

3.12.1.7 Revision of RCS Lines from Not Susceptible to TASCS to Susceptible to TASCS

The staff reviewed the applicant's proposed change related to piping lines susceptible to TASCS in Section 3.9.3.1.2 of DCD Revision 17. The staff reviewed piping and instrument drawings for these lines identified by the applicant and determined these lines are susceptible to TASCS. Therefore, the staff finds this acceptable.

3.12.1.8 Piping Design Methods

The staff reviewed the applicant's proposed change related to piping design methods and criteria in Section 3.9.3.1.5 of DCD Revision 17. The applicant summarized the methods and criteria used in design and analysis of the ASME Code Classes 1, 2, and 3 in Table 3.9-19. The staff reviewed Table 3.9-19 and determined that the applicant's summarization is acceptable.

3.12.1.9 Pipe Support Deflection Limit and Pipe Support Stiffness

The staff reviewed the applicant's proposed change related to pipe support design in Section 3.9.3.4 of DCD Revision 17. The applicant's change from dynamic loading to dynamic combined faulted loading is for clarification and considered an editorial change. The editorial change for support stiffness also provides clarification. The staff finds these editorial changes for clarification acceptable.

3.12.2 Conclusion

Based on its review of the information provided in the AP1000 Amendment, the staff concludes that supports of piping systems important to safety are designed to quality standards commensurate with their importance to safety. Section 3.12.1.1 of this report discusses the path to completion of piping analysis, including piping supports. The staff also concludes that the applicant satisfies the following:

- The requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with general engineering practice.

- The requirements of GDC 2 and GDE 4 by designing and constructing the safety-related pipe supports to withstand the effects of normal operation, as well as postulated events such as LOCAs and dynamic effects resulting from the SSE.

- 10 CFR Part 50 requirements by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service conditions to assure adequate design of all safety-related piping and pipe supports in the AP600 for their safety functions.

- 10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. Through the performance of the ITAAC, the COL holder will verify the implementation of these preapproved methods and satisfaction of the acceptance criteria. This will assure that the as-constructed piping systems conform to the certified design for their safety functions.

- 10 CFR Part 50, Appendix S, requirements by designing the safety-related piping systems with a reasonable assurance that they will withstand the dynamic effects of earthquakes with an appropriate combination of other loads of normal operation and postulated events with an adequate margin for ensuring their safety functions.

4. REACTOR

4.1 Introduction

In Revision 17 of the AP1000 design control document (DCD), Westinghouse Electric Company, LLC (Westinghouse or the applicant) proposed changes related to the reactor core and fuel design. In a letter dated October 31, 2006, to the U.S. Nuclear Regulatory Commission (NRC), Westinghouse submitted Westinghouse Commercial Atomic Power (WCAP)-16652-NP, Revision 0, APP-GW-GLR-059, “AP1000 Core & Fuel Design Technical Report” (technical report (TR)-18), to justify the proposed changes. Various sections of DCD Chapter 4, Revision 17, incorporate the majority of these proposed changes. The NRC staff’s evaluation of the proposed changes for each section of Chapter 4 of the AP1000 DCD is addressed in the corresponding section of this safety evaluation. This particular section describes the staff’s evaluation of the proposed changes to DCD Section 4.1, “Summary Description.”

In its letter of May 26, 2007, regarding its application to amend the AP1000 design certification (DC) rule, Westinghouse referred to the criterion of Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(a)(1)(vii)), “Finality of standard design certifications” and noted that these proposed changes contribute to increased standardization of the certification information.

4.1.1 Evaluation

General Design Criterion (GDC) 10, “Reactor Design,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” requires that the reactor core and associated coolant, control, and protection systems be designed to assure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs). GDC 13, “Instrumentation and Control,” requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. GDC 26, “Reactivity Control System Redundancy and Capability,” requires that two independent reactivity control systems of different design principles be provided to ensure that SAFDLs are not exceeded and that the reactor core can be held subcritical under cold conditions. GDC 27, “Combined Reactivity Control Systems Capability,” requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes to ensure that, under postulated accident conditions and with appropriate margins for stuck rods, the capability to cool the core is maintained. GDC 28, “Reactivity Limits,” requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents will not result in damage to the reactor coolant pressure boundary (RCPB) or sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. Various sections of Chapter 4 of the AP1000 DCD pertaining to the reactor core, fuel design, nuclear design, thermal-hydraulic design, and reactivity control system design describe compliance with these GDC.

DCD Section 4.1, “Summary Description,” describes the AP1000 reactor core and fuel design. The staff reviewed the proposed changes to DCD Section 4.1 and other sections in Chapter 4 for compliance with these GDC. In DCD Revision 17, the applicant proposed the following changes to DCD Tier 2, Section 4.1, related to the AP1000 reactor core and fuel design: (1) clarification of the fuel protective grid design and modification of the top nozzle design nomenclature; (2) modification of the gray rod cluster assembly (GRCA) design; (3) modification of the values of several parameters in DCD Section 4.1.1 and Table 4.1-1 for consistency; and (4) addition of four neutron panels on the thermal shield.

The current DCD Section 4.1 states that the top and bottom grids of the fuel assembly design do not contain mixing vanes. In DCD Revision 17, the applicant revised this sentence to state that the top and bottom grids and the protective grid do not contain mixing vanes. This clarification specifically states that the protective grid also does not contain mixing vanes. In DCD Table 4.1.1, the applicant revised the number of grids per fuel assembly to specifically identify one protective grid. This is a clarification of the design, not a design change, and is, therefore, acceptable.

In DCD Revision 17, the applicant proposed to change “integral clamp top nozzle (ICTN)” to “Westinghouse integral nozzle (WIN)” in the fuel assembly design. In TR-18, the applicant stated that the change from a reconstitutable ICTN to WIN in the fuel assembly is a nomenclature change. The WIN design is a proven enhancement to the ICTN design and is currently in use in the Westinghouse fleet. Both the ICTN and WIN designs eliminate the need for the top nozzle spring screws and spacer clamps. There is no significant design difference between the ICTN and WIN. The applicant described the design refinements of WIN, as compared to the ICTN, in a letter dated April 19, 2004, regarding the fuel criterion evaluation process (FCEP) notification of the WIN design (proprietary and nonproprietary) (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML041120331 and ML041120332). The staff finds that the applicant developed the WIN design using the fuel criteria evaluation process described in WCAP-12488. Therefore, the change as presented in DCD Revision 17 is acceptable.

In DCD Revision 17, the applicant proposed to change the GRCA design from 4 rodlets (out of 24) containing silver-indium-cadmium (Ag-In-Cd) to 12 rodlets containing Ag-In-Cd with reduced diameter, as described in DCD Section 4.2.2.3.2. The applicant revised DCD Table 4.1-1 to reflect this change. As discussed in Section 4.2 of this report, the staff finds the change to the GRCA design acceptable.

In DCD Section 4.1.1, the applicant changed the calculated core average power limit in the principal design requirement from 18.73 kilowatts per meter (kW/m) (5.71 kilowatts per foot (kW/ft)) to18.76 kW/m (5.718 kW/ft). The value of 18.73 kW/m (5.71 kW/ft) is a truncation of the actual value of 18.76 kW/m (5.718 kW/ft). DCD Tables 4.1-1 and 4.3-2 round the core average linear power to 18.77 kW/m (5.72 kW/ft). Therefore, the staff finds the change in Section 4.1.1 to 18.76 kW/m (5.718 kW/ft) acceptable. The applicant also revised several parameters in DCD Table 4.1-1. The effective reactor coolant flow area of heat transfer, the average velocity along the fuel rods, and the average mass velocity were revised slightly to be consistent with DCD Table 4.4-1 and the revised definition of core flow area and the core bypass flow described in DCD Section 4.4. Section 4.4 of this report addresses the revision to the core flow area and bypass.

The applicant revised DCD Table 4.1-1 to include four neutron panels on the thermal shield. The applicant made this revision to be consistent with the reactor internals design changes, which call for the addition of four neutron panels to reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values, along with the addition of a flow skirt in the lower reactor vessel head, as described in WCAP-16716-NP, Revision 2, “AP1000

Reactor Internals Design Changes" (TR-29). Since these changes could affect the core inlet flow distribution and the flow area and flow resistance in the reactor vessel downcomer and lower plenum, the staff requested that the applicant provide an evaluation of the impacts of these proposed changes on the analysis results of each of the transients and accidents described in DCD Chapter 15, "Accident Analyses."

In response to request for additional information (RAI)-TR29-SRSB-01, the applicant presented the results of the evaluation and safety analyses of the reactor vessel internals design changes. This evaluation also included the pressurizer changes described in TR-36, APP-GW-GLR-016, Revision 0, "AP1000 Pressurizer Design." (In its letter of July 18, 2008, in response to RAI-SRP10.3-SBPA-02, the applicant also stated that the evaluation of the limiting Chapter 15 event analyses provided in RAI-TR29-SRSB-01 included the revised main steam safety valve setpoints and capabilities.) The applicant concluded that these reactor internals changes have minimal impact on the fluid volume, metal masses, pressure drop through the reactor vessel, and the design reactor coolant system (RCS) flow rates used in the safety analyses. The applicant performed the evaluation or analyses on the limiting event of each of the event categories discussed in DCD Chapter 15. The analysis results for these events demonstrate that the applicable acceptance criteria for each event are met or that the existing analysis is bounding.

In its June 30, 2008, letter, the applicant provided APP-GW-GLE-026, Revision 0, "Application of ASTRUM Methodology for Best Estimate Large Break Loss of Coolant Accident Analysis for AP1000." Subsequently, the applicant, in a letter dated February 3, 2009, submitted APP-GW-GLE-026, Revision 1. This TR describes a reanalysis of a large-break loss-of-coolant accident (LOCA) using the NRC-approved uncertainty treatment method, Automatic Statistical Treatment of Uncertainty Method (ASTRUM). This large-break LOCA reanalysis includes the reactor vessel internals design changes discussed above. The results of this large-break LOCA analysis with the design changes to date show compliance with the acceptance criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." In the review of large-break LOCAs, Section 15.2.6.5 of this report addresses the staff evaluation of this TR.

DCD Section 4.1.1, "Principal Design Requirements," lists the criteria, or principal design requirements, that must be met by the mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems. These principal design criteria are designated Tier 2* information, which indicates that staff approval will be required before implementation of any change to this information. Various sections of Chapter 4 of this report describe other Tier 2* information related to the reference core design. In Section 4.1 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the staff states that the following sections in DCD Tier 2, Chapter 4, include Tier 2* information:

- WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," issued October 1994

- Principal Design Requirements

- Maximum Fuel Rod Average Burnup of 62,000 Megawatt-Days per Metric Ton of Uranium (MWDF/MTU)

- Table 4.3-1, Reactor Core Description (First Cycle)

- Table 4.3-2, Nuclear Design Parameters (First Cycle)

- Table 4.3-3, Reactivity Requirements for Rod Cluster Control Assemblies

In DCD Revision 16, the applicant proposed to reclassify all Tier 2* information throughout Chapter 4 as Tier 2 information to allow future changes to this information to be implemented without prior NRC approval. However, in response to RAI-SRP4.2-SRSB-01, the applicant stated that it would withdraw the request that Tier 2* items be reassigned to Tier 2 status in Chapter 4. Therefore, all Tier 2* items in DCD Chapter 4, including the principal design requirements, remain Tier 2* items. In DCD Revision 17, all Tier 2* information currently identified in DCD Revision 15 is restored to Tier 2*. In DCD “Introduction,” Table 1-1, “Index of AP1000 Tier 2 Information Requiring NRC Approval for Change,” these Tier 2* items, which were included in DCD Revision 15 but deleted in Revision 16, have been restored. The staff finds this acceptable.

4.1.2 Conclusion

Based on this evaluation, the staff concludes that the changes to DCD Section 4.1 continue to meet the requirement of GDCs 10, 13, 26 and 28 as described above. In addition, the results of the large break LOCA analysis with these design changes comply with the acceptance criteria of 10 CFR 50.46. Therefore, the proposed changes are acceptable.

4.2 Fuel System Design

Revision 17 of the AP1000 DCD includes the following changes to Section 4.2 (as compared to Revision 15): (1) reference changes; (2) densification value; (3) control rod descriptions; (4) burnable absorber rod design; (5) debris protection package description; (6) top nozzle nomenclature; and (7) grid fabrication description.

The staff based its review of the AP1000 fuel design on the information in the DCD and the TRs referenced by the applicant. The review was limited in scope to the changes to DCD Revision 15, as presented in Revision 17. The staff conducted its review in accordance with the guidelines provided in NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 4.2, “Fuel System Design,” which prescribes acceptance criteria to ensure that certain requirements of 10 CFR Part 50 are met. In particular, the AP1000 fuel design must meet the following GDC:

- GDC 10
- GDC 27
- GDC 35, “Emergency Core Cooling”

The fuel design must also meet the requirements of 10 CFR Part 100, “Reactor site criteria.” Thus, in reviewing the AP1000 fuel system design, the staff’s objective was to ensure that the design fulfills the following criteria:

- The fuel system will not be damaged during any condition of normal operation, including the effects of AOOs.

- Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when required.
- The number of fuel rod failures for postulated accidents is not underestimated.
- Coolability is always maintained.

The term “will not be damaged” means that the fuel rods will not fail, the fuel system’s dimensions will remain within operational tolerances, and the functional capabilities will not be reduced below those assumed in the safety analysis. These objectives address GDC 10, and the design limits that accomplish these objectives are referred to as SAFDLs. In a “fuel rod failure,” the fuel rod leaks and the first fission product barrier (i.e., the fuel cladding) is breached. The applicant must account for fuel rod failure in the dose analysis for postulated accidents required by 10 CFR Part 100. The radiological dose consequences criteria given in 10 CFR 50.34(a)(1), “Contents of applications; technical information,” are referenced in 10 CFR 100.21, “Non-seismic siting criteria.” Compliance with dose consequence criteria in 10 CFR 50.34(a)(1), with the site parameters postulated for the design, is discussed in Section 15.3 of this report.

“Coolability,” which is sometimes termed “coolable geometry,” is the ability of the fuel assembly to retain the geometrical configuration of its rod bundle with adequate coolant channel spacing for removal of residual heat. GDC 27 and GDC 35 specify the general requirements for maintaining control rod insertability and core coolability. In addition, 10 CFR 50.46 establishes specific requirements for the performance of the ECCS following postulated LOCAs. Compliance with 10 CFR 50.46 is discussed in Section 15.2.6.5 of this report.

4.2.1 Evaluation

The applicant made various reference changes in Revision 17. When needed, the applicant included additional references. In other cases, the applicant replaced references to correct erroneous references found in Revision 15. For example, the applicant deleted an erroneous reference for the integral fuel burnable absorber design in Section 4.2.2.1. In RAI-SRP4.2-SRSB-04, the staff requested that the applicant include the correct reference instead of simply removing the incorrect reference. In its response, the applicant committed to add a reference to WCAP-12610-P-A to the conclusion of Section 4.2.2.1. DCD Section 4.2.6 includes this report as Reference 5. The staff has reviewed these changes and finds them acceptable.

Throughout Section 4.2, the applicant revised the description of the control rod assemblies to clarify that the control rod clusters consist of both rod cluster control assemblies (RCCA) and gray rod cluster assemblies (GRCA). The term gray rod refers to clusters of neutron absorber rods with reduced rod worth relative to that of an RCCA. The GRCAs are used in load follow maneuvering to provide a mechanical shim reactivity mechanism to eliminate the need for changes to the concentration of soluble boron. Additionally, the applicant changed the GRCA design. Specifically, the applicant increased the number of rodlets from 4 to 12 and correspondingly reduced the diameter, such that the overall worth of the GRCA is essentially the same, as explained in the applicant’s response to RAI-SRP4.2-SRSB-05, but the assembly power profile is more even. The change in the GRCA provides a more distributed absorber material within the assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbation. DCD Revision 17 does not present the impact of these GRCAs on determining the worst-case scenarios for accident analyses. The applicant's response to RAI-SRP4.2-SRSB-05 indicated that the Chapter 15 analyses remain bounding because the use of the newer GRCA design results in a more uniform rod power throughout the affected assemblies and does not affect the worst-case assumptions used in the accident analyses. The NRC staff found this acceptable based on the RAI response and the nonsafety status of the GRCAs.

The applicant changed the burnable absorber rod design from wet annular burnable absorbers (WABAs) to borosilicate glass. However, the reference for the borosilicate design presented (WCAP-7113) is from 1967 and does not appear to have been previously approved by the NRC.

Additionally, the DCD continues to list the WABA design as an alternative. The staff issued RAI-SRP4.2-SRSB-02, RAI-SRP4.2-SRSB-03, and RAI-SRP4.2-SRSB-07 regarding these items. The applicant's response pointed out previous NRC approval of Reference Safety Analysis Reports (RESARs), that used the same reference (WCAP-7113), covering borosilicate glass RCCAs, but the applicant did not provide a specific RESAR reference. The applicant also referred to operational experience with the borosilicate glass rodlets, including a single failure of a rodlet that became detached and remained in the guide tube. In response to RAI-SRP4.2-SRSB-07, the applicant states that borosilicate burnable absorbers were used previously in a Westinghouse core design with a 14-ft core and a feed fuel enrichment of 4.4 percent. This is very similar to the AP1000 core. The response further explained that there are no changes to rodlet diameter and no changes to boron concentration for the AP1000 design as compared with the previously approved design. Based on the responses to RAI-SRP4.2-SRSB-02, RAI-SRP4.2-SRSB03, and RAI-SRP4.2-SRSB-07, the staff concludes that the use of borosilicate glass burnable absorbers is acceptable.

The applicant provided further description of the debris mitigation package in DCD Sections 4.2.2 and 4.2.2.1. Specifically, the applicant explicitly listed the parts of the debris mitigation package that include a zirconium oxide coating on the bottom section of the fuel cladding.

The applicant changed the top nozzle design in Section 4.2 (and other sections) from "Integral Clamp Top Nozzle (ICTN)" to "Westinghouse Integral Nozzle (WIN)." As described in Section 4.1.1 of this safety evaluation report (SER), the staff found this change to be acceptable.

The applicant further explained the grid fabrication process in Section 4.2.2.2.4. This information did not appear to change the grid design from that presented in DCD Revision 15 and is, therefore, acceptable.

In accordance with TR-119, APP-GW-GLR-119, "AP1000 Design Control Document Chapter 4 Tier 2* Information," the applicant edited DCD Revision 16 to remove Tier 2* designation from various fuel-related items. The applicant argued, in TR-119, that these items would be provided in a future "core reference report" and as such they could be reclassified as Tier 2. In RAI-SRP4.2-SRSB-01, the staff expressed concern that the reclassification could theoretically be interpreted to allow changes to the affected items after the core reference report was approved. In response, the applicant stated that it would withdraw the request for reclassification of all items presented in TR-119 and restore, in DCD Revision 17, all Tier 2* information currently identified in DCD Revision 15. The staff verified that Section 4.2 of Revision 17 includes all of the items designated as Tier 2* in Revision 15. In addition, Revision 17 for Table 1-1 of the DCD "Introduction" also includes these Tier 2* items, which require NRC approval for change. This staff finds this acceptable.

In Revision 15 of the DCD, Sections 4.2.5, 4.3.4, and 4.4.7 state that combined license (COL) applicants referencing the AP1000 certified design will address changes to the reference design presented in the DCD to the fuel burnable absorber rods, RCCAs, or initial core design. DCD Tier 2, Table 1.8-2, lists these COL applicant actions as COL Information Items 4.2-1, 4.3-1, and 4.4-1. In DCD Revision 17, the applicant proposed to revise Sections 4.2.5, 4.3.4, and 4.4.7.1 to state that: (1) APP-GW-GLR-059 completely addressed the COL information requested in these sections; (2) the DCD incorporated applicable changes; and (3) no additional work is required by the COL applicant to address the COL information requested in this section. In DCD Revision 17, the applicant revised Table 1.8-2 to require no COL applicant action for COL Information Items 4.2-1, 4.3-1, and 4.4-1. These COL information items intend for the COL applicant to provide information regarding the changes to the referenced reactor core design in the AP1000 DC. To increase standardization of the certified design, the applicant submitted APP-GW-GLR-059, which provides information that addresses changes to the reference design of the fuel burnable absorber rods, RCCAs, or initial core design in Revision 15 of DCD. DCD Revision 17 incorporates the applicable changes.

Based on the evaluation discussed in this report, the staff concludes that COL Information Items 4.2-1, 4.3-1, and 4.4-1 associated with DCD Revision 15 have been completed, and Revision 17 of DCD Sections 4.2.5, 4.3.4, and 4.4.7.1 is acceptable. It should be noted that the referenced design of the fuel burnable absorber rods, RCCAs, and initial core design parameters described in DCD Chapter 4 are classified as Tier 2* information, as described in Section 4.1.1. It is likely that future advancements in the design of these items could occur and COL applicants might desire these improved designs. Any change to the Tier 2* information will require prior approval by the NRC.

4.2.2 Conclusion

The staff concludes that the AP1000 fuel system, as defined by the DCD Revision 17 changes to the approved design, has been designed so that: (1) the fuel system will not be damaged as a result of normal operation and AOOs; (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required; and (3) core coolability will always be maintained, even after severe postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35; and 10 CFR Part 50.34.

The applicant provided sufficient information indicating that the changes to the approved design, as detailed in the DCD, meet the guidance provided in NUREG-0800 Section 4.2.

4.3 Nuclear Design

The staff reviewed Section 4.3 in Revision 17 of the AP1000 DCD. The staff conducted its evaluation in accordance with the guidelines provided in NUREG-0800 Section 4.3, "Nuclear Design."

In DCD Revision 17, Section 4.3, "Nuclear Design," the applicant proposed changes to the following areas related to the AP1000 nuclear design: (1) online monitoring of power distribution; (2) gray rod assembly design; (3) criticality design method outside the reactor, including soluble boron credit methodology; (4) deletion of a specific value for the moderator temperature change that accounts for the control system deadband; and (5) typical control bank worth for the initial cycle. In support of these changes, the applicant submitted APP-GW-GLR-059. The applicant originally proposed in DCD Revision 16 to reclassify the Tier 2* information in DCD Section 4.3 to Tier 2. However, in DCD Revision 17, the applicant rescinded this proposal and restored this information to its original Tier 2* designation.

4.3.1 Evaluation

DCD Tier 2, Section 4.3, “Nuclear Design,” presents the design bases for the AP1000 nuclear design. The nuclear design must ensure that the SAFDLs will not be exceeded during normal operation, including AOOs, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core.

Section 4.3 of NUREG-0800 outlines relevant requirements of Commission regulations for this area of review and the associated acceptance criteria, which include the following:

- GDC 13 requires a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

- GDC 26 requires, in part, a reactivity control system capable of holding the reactor subcritical under cold conditions.

- GDC 28 requires, in part, that the reactivity control systems be designed to limit reactivity accidents so that the RCS boundary is not damaged beyond limited local yielding.

In Section 4.3.2.2.6, the applicant modified the text to state, “Online monitoring system is not a required element for a short term reactor operation,” and added the statement, “Limits are placed on the axial flux difference so that the heat flux hot channel factor (FQ) is maintained within acceptable limits.” In Section 4.3.2.2.9, the applicant modified the text to state, “the in-core and ex-core detector systems provide adequate monitoring of power distributions when the online monitoring system is out of service.”

During the review, the staff requested additional information on why online monitoring system outage is not a required element for a short-term operation. In response to RAI-SRP4.3-SRSB-01, the applicant stated the following:

> in the unlikely event that the Online Power Distribution Monitoring System (OPDMS) should become inoperable, reactor operation can continue with shutdown margin and power distribution controls established by bounding analyses and implemented by Technical Specification Limiting Conditions for Operations (LCOs) 3.1.5, 3.1.6, 3.2.1, 3.2.2, 3.2.3 and 3.2.4. These LCOs become applicable immediately or as otherwise specified in the associated Technical Specification applicability statements, when the OPDMS is inoperable.

The staff agrees with the RAI response and finds the proposed change acceptable. In Section 4.3.2.2.6 of the DCD, the phrase “short-term” refers to the time needed to restore the online power distribution monitoring system to operable status.

In the event that the online monitoring system is out of service for a short time, the in-core and ex-core detectors provide the operator with the necessary information regarding the power distribution based on the bounding and precalculated analysis. Therefore, the online monitoring system is not a required element. The staff finds the proposed change acceptable. The applicant provided a clarification change to Sections 4.3.2.2.4, 4.3.2.2.6, and 4.3.2.2.9 regarding the axial, limiting power distribution, and monitoring instrumentation. The clarification states that, in the event the online monitoring system is out of service, limits placed on axial flux differences are designed so that the heat flux hot channel factor and departure from nuclear boiling ratio (DNBR) are maintained within acceptable limits. The staff finds this clarification to be acceptable.

In Section 4.3.2.4.13, the applicant proposed to change the number of GRCA Ag-In-Cd rodlets from 4 to 12 reduced-diameter rodlets and to change the stainless steel (SS) rodlets from 20 to 12. The total number of rodlets is unchanged at 24. The applicant made a related change to the description of the GRCAs in DCD Section 4.3.2.2.2, Figure 4.3-8 and Figure 4.3-11. Specifically, the applicant changed the gray bank from M0 to MA+MB. This change also applies to the corrections made to the titles of Figures 4.3-8 and 4.3-11. In Section 4.3.2.4.16, the applicant added the statement, “Gray rod operation is a Condition I event which includes the periodic exchange of gray rod banks.”

The term “gray rod” refers to the reduced reactivity worth relative to that of an RCCA. The GRCAs are used in load follow maneuvering to provide a mechanical shim reactivity mechanism to eliminate the need for changes to the concentration of soluble boron. The change in the GRCA provides a more distributed absorber material within the assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbation.

This change affects information in Sections 4.3 and 4.3.2.4.13, as well as Table 4.3.1. This change is not expected to impact the conclusion of Chapter 15 and is, therefore, acceptable. Condition I comprises normal operation and operational transients that are accommodated with margin between any plant parameter and the value of that parameter requiring either automatic or manual protective action. The staff agrees with categorizing GRCA operation as a Condition I event and finds the proposed addition of the statement, “gray rod operation is a Condition I event which includes the periodic exchange of gray banks,” to Section 4.3.2.4.16 acceptable.

In Section 4.3.2.6.1, “Criticality Design Method Outside the Reactor,” the applicant deleted the formula describing the total uncertainty of the criticality calculation, and added the soluble boron credit methodology to Section 4.3.2.6.2. The methodology used in soluble boron credit analysis references WCAP-14416-P (Reference 63). This WCAP is not approved by the NRC; therefore, this proposed change is not acceptable and requires further review by the NRC staff. The staff identified the issue in RAI-SRP4.3-SRSB-03. Since this issue was also identified in RAI-SRP9.1.1-SRSB-05 pertaining to the spent fuel storage rack criticality analysis, the staff designated this as Open Item OI-SRP9.1.1-SRSB-01. In response to RAI-SRP9.1.1-SRSB-05, which also serves as the response to RAI-SRP4.3-SRSB-03, the applicant indicated that it has completely revised the AP1000 spent fuel pool (SFP) criticality analysis in APP-GW-GLR-029, Revision 0, “Spent Fuel Storage Racks Criticality Analysis,” with a new methodology that meets 10 CFR 50.68 requirements. The new analysis is APP-GW-GLR-029, Revision 1, “AP1000 Spent Fuel Racks Criticality Analysis” which is a complete rewrite to supersede Revision 0. In addition the applicant has stated that the new method and models replace specific shortcomings that the NRC identified as being no longer reliable as “approved methodology.” In a letter dated September 29, 2009, the applicant submitted APP-GW-GLR-029, Revision 2, which is identical to Revision 1, except for minor updates to include pyrex insert burnable absorber design among other burnable absorbers. These new efforts include evaluation of soluble boron. The applicant also proposed to revise DCD Sections 4.3.2.6.1 and 4.3.2.6.2 to be consistent with the methodology described in APP-GW-GLR-029, Revision 2. The staff evaluation of spent fuel criticality analysis of APP-GW-GLR-029, Revision 2, is addressed in Section 9.1.2.2.4 of this report. The staff confirmed that DCD Sections 4.3.2.6.1 and 4.3.2.6.2 were revised in a subsequent revision to the DCD.

In Section 4.3.2.4.2, the applicant proposed to delete the specific value of the Δ2.22 °Celsius (C) (Δ4 °Fahrenheit (F)) moderator temperature increase that accounts for the control system deadband. The shutdown margin control requirements calculation accounts for the control system deadband and measurement uncertainties by assuming that the moderator temperature is at its maximum possible value before plant trip. This conservatively increases the change in moderator temperature when going from hot full power to hot zero power after plant trip, thereby increasing the shutdown margin control requirement. The uncertainty is based on conservative engineering judgment and includes both instrument errors and deadband. The AP1000 analyses used a preliminary T_{avg} uncertainty, which is provided in the DCD. The applicant revised Section 4.3.2.4.2 to reflect that the allowance for deadband and measurement errors is not set at a fixed value of Δ2.22 °C (Δ4 °F). This change does not affect the conclusion of Chapter 15; therefore, it is acceptable.

The applicant revised Tables 4.3-1 and 4.3-2 and Sections 4.3.2.2.7, 4.3.4, and 4.3.5 to reflect the DCD Chapter 15 accident analysis input assumptions and its results. The applicant revised the fuel assemblies' diameter of guide thimbles and lower part to be consistent with the other dimension values given in the table. The applicant revised the nuclear design parameters, reactivity coefficient, and doppler coefficients in Table 4.3-2 to correctly refer to Figure 15.0.4-1 and to be consistent with the DCD Chapter 15 accident analysis. The applicant revised Section 4.3.4 to reflect COL information updates. The applicant also updated Sections 4.3.2.2.7 and 4.3.5 to correct, delete, and add new references. These changes provide consistency with other DCD sections and do not impact the conclusion of Section 4.3 or Chapter 15; therefore, they are acceptable.

In Table 4.3-2, the applicant provided first-cycle values for the typical hot channel factors and bank worth from beginning of life to end of life. The changes to these values represent first-cycle updated results of the in-core fuel management scheme and are acceptable.

In Section 4.3.2.4.16, the applicant revised mechanical shim load follow and base load operations (including the gray rod bank insertion sequence exchanges) to establish a more negative value than the axial offset associated with all-rods-out condition. The staff considers this to be a conservative change and it is, therefore, acceptable.

The applicant revised Section 4.3.4 to state that APP-GW-GLR-059 (Reference 64) completely addresses the COL information requested in this section, and the DCD incorporates applicable changes. No additional work is required by the COL applicant to address the COL information requested in this section. This change is acceptable as discussed in Section 4.2 of this report.

Revision 17 of DCD Section 4.3 includes numerous editorial and clarification changes concerning specifications that are covered in the core operating limits report, general formatting changes to the references, and clarifications to the variable units used in Section 4.3 of the DCD. The staff finds these clarifications to be acceptable.

4.3.2 Conclusion

The staff reviewed the changes to DCD Revision 17 that demonstrate that sufficient control rod and burnable poison worth exist to provide safe shutdown of the plant. Furthermore, the control rod system is designed to ensure that reactivity accidents do not result in damage to the RCPB. Therefore, the staff finds that the proposed DCD changes comply with the requirements of GDC 13, 26, and 28 and are acceptable. The staff expects the above changes to apply to all COL applications referencing the AP1000 DC.

4.4 Thermal-Hydraulic Design

In Revision 17 to the AP1000 DCD, the applicant proposed changes related to the reactor core thermal-hydraulic design. In a letter dated October 31, 2006, the applicant submitted TR-18, which provides rationale and justification for the proposed changes related to the reactor core and fuel design. In TR-108, APP-GW-GLN-019, "AP1000 Standard Combined License Technical Report, Fluid System Changes," the applicant proposed revisions to the AP1000 fluid systems, including the digital metal impact monitoring system (DMIMS). DCD Revision 17, Section 4.4, includes the proposed changes that affect the reactor core thermal-hydraulic design.

In its letter of May 26, 2007, regarding its application to amend the AP1000 DC rule, the applicant referred to the criterion of 10 CFR 52.63(a)(1)(vii) and stated that these proposed changes contribute to increased standardization of the certification information.

4.4.1 Evaluation

DCD Section 4.4, "Thermal and Hydraulic Design," describes the AP1000 reactor core thermal-hydraulic design to ensure adequate heat removal to prevent fuel damage during normal operation and transients. In Revision 17 of the AP1000 DCD, the applicant proposed the following changes to Section 4.4 related to the AP1000 core thermal-hydraulic design: (1) revision of the core bypass flow; (2) change of the maximum rod bow penalty for DNBR calculation from less than 1.5 percent DNBR to less than about 2 percent DNBR; (3) change of the hydraulic loads calculation from the mechanical design flow to the best estimate flow; (4) change of the peak linear power resulting from overpower transient/operator errors from 73.82 kW/m (22.5 kW/ft) to less than or equal to 73.65 kW/m (22.45 kW/ft); (5) change of the term "canned motor pump" to "reactor coolant pump"; (6) revision of the description of the DMIMS; (7) changes of the values of several parameters in Table 4.4-1; (8) addition of WCAP-15063-P-A, Revision 1; (9) addition of Reference 87 (WCAP-16652-NP); and (10) division of Section 4.4.7, "Combined License Information," into Sections 4.4.7.1 and 4.4.7.2, with revisions to each section.

GDC 10 specifies that the reactor core and the associated coolant, control, and protection systems must be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In its review of the AP1000 DCD changes related to DCD Section 4.4, the staff used the guidance of NUREG-0800 Section 4.4, "Thermal and Hydraulic Design," which sets forth the acceptance criteria used by the staff to evaluate the thermal-hydraulic design of the reactor core for compliance with the relevant requirements of GDC 10. The following discusses the evaluation of these changes.

DCD Section 4.4.1.3.1, "Design Basis," identifies the core bypass flow (which is not considered effective for heat removal) as the coolant flow through the thimble tubes and the leakage from the core barrel shroud region into the core. Revision 17 revised the core bypass flow as the coolant flow through the thimble and instrumentation tubes and the leakage between the core barrel and core shroud, head cooling flow, and leakage to the vessel outlet nozzles. In TR-18, the applicant indicated that it made this change to accurately describe the core cavity flow area caused by a change from the core baffle-former design to a welded core shroud design, which eliminates the need for bolts in the high fluence regions immediately adjacent to the reactor core. The revision to DCD Section 4.4.1.3.1 clarifies the regions of the core bypass flow that are not effective for core heat transfer. The shroud core cavity flow is considered to be active flow that is effective for fuel rod cooling. Therefore, the applicant deleted Item E in Section 4.4.4.2.1 from a core bypass flow path to be consistent with core bypass flow design basis detailed in Section 4.4.1.3.1. This revision results in slight changes in the effective core flow area, average velocity along the fuel rods, and average mass velocity, as reported in Tables 4.1-1 and 4.4-1, from 3.85 square meters (m^2) (41.5 square feet (ft^2)), 4.85 meters per second (m/s) (15.9 feet per second (ft/s)), and 3268.5 kilograms per second-square meter (kg/s-m^2) (2.41×10^6 pounds per hour-square feet) (lbm/hr-ft^2)) to 3.88 m^2 (41.8 ft^2), 4.82 m/s (15.8 ft/s), and 3458.4 kg/s-m^2 (2.55×10^6 lbm/hr-ft^2), respectively.

DCD Section 4.4.4.2.1 states that calculations using drawing tolerances in the most conservative direction and accounting for uncertainties in the pressure losses show the core bypass to be no greater than the 5.9-percent design value. The maximum value of 5.9 percent allotted as bypass flow and the thermal design flow of 94.1 percent of the thermal flow rate assumed for the core cooling evaluations, as stated in DCD Sections 4.4.1.3.2 and 4.4.1.3.1, respectively, remain unchanged. Thus, the change in the core bypass flowpaths has insignificant effects on the core cooling calculation and is, therefore, acceptable. The change in the average mass velocity reflects a correction to the coolant temperature used in the calculation of the core average flow rate from the core average temperature to the core inlet temperature. This is acceptable because the calculation with higher water density of the core inlet temperature results in a higher mass velocity with the same volumetric flow rate.

In DCD Revision 17, the applicant added Reference 88 to Sections 4.4.1.2.1 and 4.4.8 (the reference section). Reference 88 addresses the NRC approval of the maximum burnup limit of 62,000 MWD/MTU for WCAP-10444-P-A, WCAP-12610-P-A, WCAP-12488-A, and WCAP-15063-P-A, all of which are referenced in Section 4.4.8. The applicant added Reference 88 for completeness, and the staff finds this addition acceptable.

In DCD Revision 17, the applicant added Reference 82a to Sections 4.4.2.2.1 and 4.4.8. Reference 82a addresses the application of an adjustment factor to the WRB-2M critical heat flux correlation described in Reference 82, which is used in the AP1000 thermal-hydraulic design calculations. The applicant added Reference 82a for completeness, and the staff finds this addition acceptable.

Section 4.4.2.2.5 states that the maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU and indicated that the maximum rod bow penalty is less than 1.5 percent DNBR. In Revision 17, the applicant changed the maximum rod bow penalty of “less than 1.5 percent DNBR” to “less than about 2 percent DNBR.” In TR-18, the applicant stated that it made this change for consistency with the basis used for 3-D flow-accelerated corrosion (FAC) analysis. Because the change of the maximum penalty from 1.5 percent to about 2 percent DNBR is a more conservative change, and the actual value of the rod bow penalty used in the safety analysis will be calculated based on actual fuel design and assembly burnup of 24,000 MWD/MTU, the staff concludes that the change is acceptable.

Section 4.4.2.6.2 states that the hydraulic loads at normal operating conditions are calculated considering the mechanical design flow and account for the minimum core bypass flow based on manufacturing tolerances. In Revision 17, the applicant changed the mechanical design flow to the best estimate flow. As stated in TR-18, the applicant changed the mechanical design flow to the best estimate flow for the hydraulic calculation to achieve consistency with the current design procedures. Therefore, the staff finds the change to be acceptable.

In DCD Revision 17, Section 4.4.2.7.1, the applicant corrected a typographic error in the Dittus-Boelter correlation for the force convection heat transfer coefficients calculation. However, the staff found that the correction still includes editorial errors. The applicant stated, in response to a telecom of November 17, 2008, that it will correct these errors in a follow up version of DCD Section 4.4.2.7.1. In a subsequent revision to the DCD, the applicant made appropriate changes to DCD Section 4.4.2.7.1, which resolves this issue.

In Section 4.4.2.11.6, the applicant changed the peak linear power resulting from overpower transient or operator errors from 73.82 kW/m (22.5 kW/ft) to less than or equal to 73.65 kW/, (22.45 kW/ft). This change ensures consistency with the reactor design value specified in Tables 4.1-1 and 4.4-1 and is, therefore, acceptable.

In Section 4.4.4.6, the applicant changed the term “canned motor pump” to the more generic term, “reactor coolant pump.” This editorial change does not affect the negative slope for the pump head-capacity curve as a generic reactor coolant pump characteristic and is, therefore, acceptable.

In Section 4.4.6.4, the applicant made changes to the description of the DMIMS. In TR-103, APP-GW-GLN-019, Revision 2, “AP1000 Standard Combined License Technical Report, Fluid Systems Changes,” the applicant stated that it made these changes to correct a misinterpretation of the required number of sensors and to correct and delete incorrect information. The requirement for loose parts monitoring system sensors was incorrectly interpreted as requiring four, rather than two, sensors per collection region. This correction results in the removal of the term “redundancy” regarding sensors at each RCS location and instrumentation channel. The applicant made other changes to the descriptions of the DMIMS performance tests method and the technique used to minimize false impact detection. The applicant made these changes to accurately represent the DMIMS design, and they do not result in a change of the design. The DMIMS design continues to conform to system design aspects of Regulatory Guide (RG) 1.133, Revision 1, “Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors.”

In Table 4.4-1, the applicant changed the values of several parameters to be consistent with Table 4.1-1. In Footnote j, the applicant revised the theoretical density of the fuel from 95 percent to 95.5 percent for AP1000 (95 percent for others). The applicant modified Footnote f, which is associated with heat transfer, by changing the phrase, “based on 157 fuel assemblies and hot densified fuel length” to “based on densified active fuel length.” The applicant rounded the value for AP1000 to 18.77 kW/m (5.72 kW/ft). The changes regarding the theoretical density are acceptable as they are consistent with DCD Section 4.2.3.2.

In Section 4.4.2.11, the applicant added Reference 85, WCAP-15063-P-A, Revision 1, “Westinghouse Improved Performance Analysis and Design Model (PAD 4.0).” This report presents an NRC-approved fuel rod design methodology. The addition of a reference to this topical report reflects the NRC's approval of the methodology. Therefore, this addition is acceptable.

DCD Revision 16, Section 4.4, proposed to change Reference 9, WCAP-12488-A, from Tier 2* information to Tier 2. As discussed in Section 4.1 of this report, in response to RAI-SRP4.2-SRSB-01, the applicant stated that it would withdraw the request that this Tier 2* item be reassigned to Tier 2 status in Chapter 4. In DCD Revision 17, the applicant restored all Tier 2* information identified in DCD Revision 15, including WCAP-12488-A, to Tier 2*. The staff finds this acceptable.

In Revision 17, the applicant revised Section 4.4.7.1 to state that APP-GW-GLR-059 (Reference 87) completely addressed the COL information requested in this section and the DCD incorporated applicable changes; therefore, no additional work is required by the COL applicant to address the COL information requested in this section. The staff finds this revision to be acceptable, as discussed in Section 4.2 of this report.

In DCD Revision 17, Section 4.4.7.2 states the following:

> Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in Subsection 7.1.6, and prior to fuel load, the Combined License holder will calculate the design limit DNBR values. The calculations will be completed using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in Section 4.4, "Thermal and Hydraulic Design," remain valid, or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty.

This is a change from DCD Revision 15, Section 4.4.7, "Combined License Information." The applicant has changed the "combined license applicant" who will calculate the design limit DNBR values to the "combined license holder" who will calculate the design limit DNBR values before fuel load. In response to RAI-SRP4.4-SRSB-02, the applicant stated that the design limit DNBR depends on the selection of the actual plant operating instrumentation, as well as on calculation of the instrumentation uncertainties of the operating plant parameters. The applicant further stated the following:

> The actual calculated instrumentation uncertainties will be used, when available, to recalculate the RTDP [revised thermal design procedures] DNBR design limits. If these recalculated DNBR design limits are less than those given in DCD 4.4.1.1.2, then the DCD values are conservatively high and will not be revised. If the calculated values are higher than the values in the DCD, then the design limits will be revised. Note that since conservatively high values of the instrumentation uncertainties were used in the calculations, we do not expect the design limits to change.

The staff agrees that the actual design limit DNBR can only be calculated after the selection of actual plant operating instrumentation and, therefore, the change from "the combined license applicant" to the "combined license holder" who will calculate actual design limit DNBR values is acceptable. It should be noted that in NUREG-1793, Appendix F, "Combined License Action Items," Item 4.4-1 reiterated DCD COL Information Item 4.4-2. Therefore, this same change in COL Information Item 4.4-2, as described in DCD Section 4.4.7.2, is applied to NUREG-1793

Item 4.4-1. Also, in Revision 17 of the DCD, the applicant revised Table 1.8-2, “Summary of AP1000 Standard Plant Combined License Information Items,” to be consistent with the changes to Section 4.4.7.2 to indicate that COL Information Item 4.4-2 is an action required by the COL holder. This is acceptable to the staff.

4.4.2 Conclusion

The staff has reviewed the changes to DCD Section 4.4. Based on the evaluation discussed above, the staff concludes that the core thermal-hydraulic design continues to meet the requirements of GDC 10 and is, therefore, acceptable.

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

4.5.1.2.1 Material Specification

In Revision 16 to the AP1000 DCD, the applicant proposed changes to Section 4.5.1 to include austenitic SS Types 304, 304L, 316, and 316L for parts of the control rod drive mechanisms (CRDMs) and control rod drive (CRD) line exposed to reactor coolant, including pressure-boundary components. TR-33, APP-GW-GL-009, “Pressure Boundary Material Change,” Revision 1, submitted in a letter dated May 24, 2007, identified and justified these changes.

In addition, the applicant revised DCD Tier 2, Section 4.5.1, to include the use of a cobalt alloy or qualified substitute to fabricate CRDM latches and links. The applicant also modified the nickel-chromium-iron alloy (Alloy 750) specification used for CRDM springs from Aerospace Material Specification (AMS) 5698E and AMS 5699E to AMS 5698 and AMS 5699. TR-106, APP-GW-GLN-106, “AP1000 Licensing Design Changes for Mechanical Systems and Component Design Updates,” Revision 1, submitted in a letter dated September 28, 2007, identified and justified the aforementioned changes.

4.5.1.2.2 Evaluation

4.5.1.2.2.2 Changes to Control Rod Drive Mechanism Stainless Steel Materials (TR-33)

GDC 1, “Quality Standards and Records,” requires structures, systems, and components (SSCs) important to safety to be designed, fabricated, erected, and tested to quality standards to ensure a quality product in keeping with the required safety function. GDC 4, “Environmental and Missile Dynamic Effects Design Bases,” requires that SSCs important to safety be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff review of the AP1000 DCD changes related to this section ensures that the materials for the CRD system meet the American Society of Mechanical Engineers (ASME) Code requirements and are compatible with the reactor coolant environment to ensure a quality product commensurate with its importance to safety.

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the CRDM and CRD line. The parts of the CRDMs and CRD line exposed to reactor coolant are made of materials designed to resist the degradation mechanisms of the reactor environment.

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the CRDM. The parts of the CRDM exposed to reactor coolant are made of materials designed to resist the degradation mechanisms of the reactor environment. Currently, DCD Section 4.5.1.1 and the corresponding Table 5.2-1 in Section 5.2.3 include SS Types 304LN and 316LN, which have high resistance to sensitization. Therefore, these materials are more resistant to stress-corrosion cracking (SCC) because of their low carbon content. The applicant revised DCD Tier 2, Section 4.5.1, to include austenitic SS Types 304, 304L, 316, and 316L, as discussed in TR-33. TR-33, which provides the basis for the change, states that the addition of these materials will enhance manufacturing flexibility, reduce costs, and reduce risk relative to material availability. The applicant submitted changes to the AP1000 DCD, as proposed in TR-33, pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that the proposed changes contribute to increased standardization of the certification information.

SS Types 304 and 316 (higher carbon content) are less resistant to sensitization due to heat treatment or welding. In addition, NRC Information Notice 2006-27, "Circumferential Cracking in Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors," and numerous requests for relief from the ASME Code concerning repairs to leaking CRDM canopy seal welds discuss emerging issues involving SS Types 304 and 316. These instances of SCC are occurring in stagnant or dead end pressurized-water reactor (PWR) coolant environments. Since Type 304 and 316 SS materials are more susceptible to intergranular stress-corrosion cracking (IGSCC) and transgranular stress-corrosion cracking than the low-carbon SS Types 304L, 304LN, 316L, and 316LN, the use of Types 304 and 316 materials may affect the integrity of the CRD components (including the RCPB portions of the latch housing and rod travel housing). Specifically, the use of these materials can affect the structural integrity of CRD components that are subjected to stagnant water (trapped oxygen), dead legs, or areas prone to increased levels of oxygen.

Therefore, the staff requested, in RAI–SRP4.5.1-CIB1-01, that the applicant delete the proposed addition of SS Types 304 and 316 from the AP1000 DCD or provide further justification addressing the acceptability of the proposed addition of these materials.

In a letter dated May 30, 2008, the applicant provided further justification concerning the acceptability of the proposed addition of SS Types 304 and 316 in components subjected to stagnant water. The applicant stated that all austenitic SSs are procured in the solution annealed condition and that controls are established in Section 5.2.3.4 of the AP1000 DCD to avoid sensitization due to heat treatment or welding and prevent susceptibility to intergranular attack as addressed by RG 1.44, "Control of the Use of Sensitized Stainless Steel," issued May 1973. In addition, cleaning procedures and contamination preventative measures are implemented to prevent the presence of detrimental impurities that could contribute to SCC. The AP1000 CRDM design uses only one canopy seal (CRDM to the rod travel housing) in lieu of the three canopy seals in current designs. In addition, this canopy seal weld was redesigned with a larger radius and thicker wall to reduce stress levels, and the design includes a vent and drainpath for the canopy seal volume to prevent fully stagnant conditions. The number of occurrences of cracking in SS components exposed to PWR operating environment is small considering the number of SS components used in PWR applications. Most of these failures have occurred as a result of adverse conditions (trapped oxygen in stagnant water) and high stresses.

The staff agrees that using the guidance in RG 1.44 and proper cleaning techniques limits the amount of sensitization of Types 304 and 316 SS, which in turn reduces the susceptibility to SCC. In addition, the redesign of the CRDM reduces the stresses and adverse environment (vent and drainlines to minimize the presence of an oxygenated environment) that is a major contributor to the susceptibility of the SS to SCC. Therefore, the staff finds that Types 304 and 316 SS can be used, in addition to Types 304L, 304LN, 316L, and 316LN (which are less susceptible to sensitization) previously approved by the staff in NUREG-1793 because the applicant's design changes will reduce the number of canopy seal welds to one per CRDM, reduce the stresses in the canopy seal weld, eliminate the presence of an oxygenated environment using vent and drainlines, and follow the guidance in RG 1.44. However, these design changes (reduced number of welds, reduced stresses, and use of vent and drainlines) and the use of RG 1.44 are critical in preventing the occurrence of SCC in these components. However, the staff concluded that the applicant should add these design changes and the use of RG 1.44 for the CRDM components, including the canopy seal welds, to the DCD. It should be noted that DCD Section 4.5.1.2 applies the controls on preventing SCC, including the guidance in RG 1.44 for the austenitic SS pressure-housing components of the CRDM. The canopy seal welds may not be considered a pressure-housing component since they only provide a leakage barrier. The staff identified this issue as Open Item OI-SRP4.5.1-CIB1-01. In a letter dated May 13, 2009, the applicant provided a marked-up copy of the DCD, which included the design changes and the use of RG 1.44 as requested by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The identified added materials and the additional information on grades, types, or classes meet the requirements of GDC 1, GDC 4, and the ASME Code, Sections II and III, in accordance with the guidance of NUREG-0800 Section 4.5.1. Therefore, the addition of the identified materials and the additional information on grades, types, or classes presented in TR-33, Revision 1, is acceptable.

The staff reviewed the proposed material changes to the AP1000 DCD. The applicant has incorporated the proposed changes, as identified in TR-33, Revision 1, into the AP1000 DCD. Therefore, the staff finds that the DCD changes that address material changes, as proposed by the applicant in TR-33, Revision 1, are acceptable.

The proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD and are acceptable.

4.5.1.2.2.3 Changes to Control Rod Drive Mechanism Latches, Links, and Springs (TR-106)

GDC 26 requires that one of the reactivity control systems use control rods, preferably with a positive means for inserting the rods, and be capable of reliably controlling reactivity changes for assurance that fuel design limits are not exceeded under conditions of normal operation, including AOOs.

The staff reviewed the AP1000 DCD changes related to this section to ensure that the materials for the CRDMs will perform adequately throughout the design life of the plant. The CRDM latches, links, and springs are not RCPB components and are, therefore, not required to be fabricated from materials meeting ASME Code, Section III. DCD Tier 2, Section 4.5.1, describes the materials used to fabricate CRDM components.

The applicant revised DCD Tier 2, Section 4.5.1, to include a cobalt alloy or qualified substitute to fabricate CRDM latches and links. Previously, the applicant intended to fabricate latches and links from SS hard faced with Stellite.™ The applicant also deleted its requirement that all cobalt alloy material be ordered in the solution-treated, cold-worked condition. In addition, the applicant modified the nickel-chromium-iron alloy (Alloy 750) specification used for CRDM springs. The applicant also changed the specification from AMS 5698E and AMS 5699E to AMS 5698 and AMS 5699.

In TR-106, the applicant stated that the previous design of the CRDM in Section 4.5 of Revision 15 of the AP1000 DCD was based on a design utilized in past Westinghouse plants to achieve a design life of 3 million steps. The applicant modified the design to meet a design life of 8 million steps, which required a material change for CRDM latches and links. The redesigned latches have also been modified to a double tooth design constructed of solid Stellite™ in lieu of the previous design, which utilized a single tooth design and was constructed of SS hard faced with Stellite. ™

In response to RAI-TR106-CIB1-02, dated October 5, 2007, the applicant provided additional information regarding the use of solid Stellite™ components in PWR CRDMs. The applicant stated that cast single-tooth latches constructed from solid Stellite™ have been used extensively in the Combustion Engineering (CE) fleet and all CE units in Korea. CE has used single-tooth cast Stellite™ to construct latches for the past 30 years in its units. Electricite de France (EDF) has used double-tooth hardfaced latches. The AP1000 cast latches will be based on AMS 5387.

AMS 5387 (similar to Stellite™ 6) is a cast cobalt-based alloy that provides adequate resistance to wear and corrosion in the reactor coolant environment. The staff is unaware of any failures or degradation issues associated with the use of cast Stellite™ in CRDM components. Based on the corrosion resistance and mechanical properties of this material, as well as the favorable operating experience in currently operating plants, the staff finds the applicant’s use of AMS 5387 to fabricate latches and links acceptable. The applicant’s elimination of required solution heat treatment and cold-working for cobalt alloys used in the CRDMs is acceptable because the AMS 5387 used for latches and links requires that components fabricated to this specification are used in the as-cast condition. It is the staff’s understanding that cobalt alloy link pins will still be delivered in the solution treated, cold-worked condition because these components are not cast.

In addition to the modifications to DCD Tier 2, Section 4.5.1.3, discussed above, the applicant changed the specification for CRDM springs (Alloy 750) from AMS 5698E or AMS 5699E to AMS 5698 or AMS 5699. This modification does not change the materials that the staff previously approved. The modification to specifications listed for CRDM springs allows current versions of AMS 5698 and AMS 5699 to be used and is, therefore, acceptable.

The staff finds that the proposed modifications are acceptable and meet the requirements of GDC 26 and the acceptance criteria of NUREG-0800 Section 4.5.1. In addition, the NRC staff reviewed the proposed changes as they relate to Revision 16 of the AP1000 DCD. Revision 16 of the DCD incorporates the proposed changes identified in TR-106. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

4.5.1.2.3 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to DCD Section 4.5.1 meet the requirements of 10 CFR 50.55a, “Codes and standards,” and

ASME Code, Section III, and are, therefore, acceptable. The proposed changes have been incorporated into the AP1000 DCD. Furthermore, the staff finds that the TR-33, Revision 1, conclusions regarding the evaluation of the changes to the material specification are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information. In addition, based on the above evaluation, the staff finds that the revisions proposed by the applicant to AP1000 DCD, Section 4.5.1, meet the requirements of GDC 26 and, therefore, are acceptable. The AP1000 DCD incorporates the proposed changes, as identified in TR-106, Revision 1. Furthermore, the staff finds that the TR-106, Revision 0, conclusions regarding the evaluation of the changes to the material specification are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

4.5.2 Reactor Internal and Core Support Materials

In Revision 16 of the AP1000 DCD, the applicant proposed design changes related to the material specifications for AP1000 reactor internal components. This resulted in changes to Section 4.5.2.1. In a letter dated July 31, 2007, the applicant submitted TR-31, APP-GW-GLN-015, “Reactor Internals Material Changes for the AP1000 Plant,” Revision 1, which included WCAP-16624-P, Revision 1, “Reactor Internals Materials Changes for the AP1000 Plant,” to provide the technical justification for the proposed design changes related to the material specifications for reactor internal components.

In TR-31, the applicant proposed the following changes to the AP1000 reactor internal and core support material specifications: (1) the addition of American Iron and Steel Institute Types 304, 304H, and 304L SSs to the material specifications for reactor internal core structure components; (2) the use of nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts in place of strain-hardened Type 316 SS; (3) the use of nickel-based Alloy 690 for the clevis inserts in place of 304-series SSs; (4) the addition of Stellite™ 6 to the hardfacing materials for the radial keys, clevis inserts, and alignment pins; and (5) the use of nickel-based Alloy 750 for the irradiation specimen springs in place of Type 302 SS.

Other related changes identified in TR-31 include: (1) an additional specification in DCD Section 4.5.2.1 for qualification of welding procedures in accordance with the staff guidance in RG 1.44; (2) revisions to the language in DCD Section 4.5.2.1 pertaining to the susceptibility of reactor internal components to irradiation-assisted stress-corrosion cracking (IASCC) and void swelling; and (3) an additional statement in DCD Section 4.5.2.1 indicating that Alloy 600 would not be used in the AP1000 reactor internal components.

Revision 16 to DCD Section 4.5.2.1 implements all of the above changes proposed in TR-31 as follows:

> The major core support structure material for the reactor internals is SA-182, SA-479, or SA-240 Types 304, 304L, 304LN, or 304H stainless steels. Fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with RG 1.44. For threaded structural fasteners the material used is strain hardened Type 316 stainless steel and for the clevis insert-to-vessel bolts either Unified Numbering System for Metals and

Alloy UNS N07718 or N07750. Remaining internals parts not fabricated from Types 304, 304L, 304LN, or 304H stainless steels typically include wear surfaces such as hardfacing on the radial keys, clevis inserts, alignment pins (Stellite^{TM} 6 or 156 or low cobalt hardfaces); dowel pins (Type 316); hold down spring (Type 403 stainless steel (modified)); clevis inserts (UNS N06690); and irradiation specimen springs (UNS N07750). Core support structure and threaded structural fastener materials are specified in the ASME Code, Section III, Appendix I as supplemented by Code Cases N-60 and N-4. The qualification of cobalt free wear resistant alloys for use in reactor coolant is addressed in Subsection 4.5.1.3.

The use of cast austenitic stainless steel (CASS) is minimized in the AP1000 reactor internals. If used, CASS will be limited in carbon (low carbon grade: L grade) and ferrite contents and will be evaluated in terms of thermal aging effects.

The estimated peak neutron fluence for the AP1000 reactor internals has been considered in the design. Susceptibility to irradiation-assisted stress corrosion cracking or void swelling in reactor internals identified in the current pressurized water reactor fleet are being addressed in reactor internals material reliability programs. The selection of materials for the AP1000 reactor internals considers information developed by these programs. Ni-Cr-Fe Alloy 600 is not used in the AP1000 reactor internals.

4.5.2.1 Evaluation

GDC 1 and 10 CFR 50.55a require that SSCs important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed. The staff's review of the AP1000 DCD changes related to this section covers material, components design, fabrication, and inspection to ensure structural integrity in compliance with 10 CFR 50.55a and GDC 1.

TR-31 proposes changes to the material specifications for the AP1000 reactor internal components. DCD Tier 2, Section 4.5.2.1, identifies the reactor internal material specifications. In Revision 15 of the DCD, the major materials of construction for the reactor internal core support structure components, excluding threaded structural fasteners, are Type 304LN SS. Revision 15 of the DCD specifies strain-hardened Type 316 SS for threaded structural fasteners. Other reactor internal materials specified in DCD Revision 15, Section 4.5.2.1, include Stellite^{TM} 156 or low cobalt hardfacing on the radial keys, clevis inserts, and alignment pins; Type 316 SS for the dowel pins; modified Type 403 SS for the holddown spring; and Type 302 SS for irradiation specimen springs.

The proposed material specification changes primarily involve the addition of Types 304, 304H, and 304L to the 304-series SSs specified for the reactor internal core support structure components. Other material changes include: (1) the use of nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts in place of strain-hardened Type 316 SS; (2) the use of nickel-based Alloy 690 for the clevis inserts in place of 304-series SSs; (3) the addition of Stellite^{TM} 6 to the above hardfacing materials for the radial keys, clevis inserts, and alignment pins; and (4) the use of nickel-based Alloy 750 for the irradiation specimen springs in place of Type 302 SS. DCD Revision 16, Section 4.5.2.1, implements the changes to these material specifications.

Section 3 of TR-31 provides a technical description and justification for the proposed material specification changes. The addition of Types 304 and 304H SSs is significant because these SSs have higher carbon content than Type 304LN. Types 304 and 304H have a maximum carbon content of 0.08 percent (by weight) and 0.10 percent, respectively; whereas, Type 304LN is a low-carbon SS with a maximum carbon content of 0.03 percent. Type 304H, in particular, is notable as a high-carbon SS with a specified minimum carbon content of 0.04 percent. None of the other SS grades specified in DCD Revision 16, Section 4.5.2.1 (Types 304, 304L, 304LN), has a specified minimum carbon content. The 304H grade was developed to ensure better resistance to high-temperature creep by maintaining at least 0.04-percent carbon. The higher allowable carbon content in Types 304 and 304H austenitic materials can potentially result in a significant degree of sensitization to intergranular corrosion and IGSCC at elevated temperatures. In the temperature range of 426.7 °C to 815.6 °C (800 °F to 1,500 °F), chromium carbides, $(Fe,Cr)_{23}C_6$, are insoluble and precipitate at grain boundaries through the diffusion of carbon. Precipitation of chromium carbides at grain boundaries result in the depletion of chromium in the surrounding matrix alloy immediately adjacent to the grain boundaries. The chromium-depleted alloy at the grain boundaries is much less corrosion resistant than the rest of the bulk alloy (i.e., away from the grain boundaries). The galvanic coupling of chromium-depleted alloy at the grain boundaries with bulk alloy in the passive state (due to undepleted chromium) can result in significant intergranular corrosion. High-carbon SSs subjected to temperatures in the range of 426.7 °C to 815.6 °C (800 °F to 1,500 °F) for a sufficient time to allow for the formation of chromium carbides at the grain boundaries are sensitized to intergranular corrosion and IGSCC. When temperatures exceed 815.6 °C (1,500 °F), the chromium carbides are soluble, and below 426.7 °C (800 °F) the diffusion rate of carbon is not sufficient to permit the formation of chromium carbides. Therefore, it is specifically in the intermediate temperature range that sensitization is a significant concern for high-carbon SSs. Welding is known to produce sensitization in weld heat affected zones (HAZs), located on either side and, at times, somewhat removed from the actual weld bead. It is in these HAZs that the welding process can produce temperatures in this intermediate range for a sufficient time to allow for carbon diffusion and the formation of chromium carbides and chromium-depleted zones at the grain boundaries. The carbon content of low-carbon SS grades (i.e., SSs with a specified maximum carbon content of 0.03 percent) is not high enough for sensitization to be a significant concern because a sufficient quantity of carbon does not exist to cause significant chromium depletion at grain boundaries within a practical timeframe for the welding process.

High-carbon austenitic SSs in nuclear reactor structural components that have become locally sensitized are potentially susceptible to IGSCC at these sensitized locations. While significant intergranular corrosion is generally not an issue for unstressed parts, the presence of tensile stresses has been known to produce IGSCC in components where sensitization has occurred. For a given high-carbon SS component, the effects of sensitization can be minimized by controlling weld parameters, such as heat input and cooling rate. RG 1.44 describes acceptable methods for controlling the processing of SSs to avoid sensitization that could lead to IGSCC. The RG specifies that, for a given material composition, welding practices should be controlled to avoid excessive sensitization of base metal HAZs adjacent to welded joints. The RG also specifies an intergranular corrosion test for the qualification of welding procedures to be used for welding SSs having a carbon content of greater than 0.03 percent. DCD Revision 16, Section 4.5.2.1, includes language requiring that welding procedures be qualified in accordance with RG 1.44.

DCD Revision 16, Section 4.5.2.1, also added Type 304L, a low-carbon grade of SS (0.03-percent maximum) similar to Type 304LN, to the list of permissible SS grades for reactor internal core support structure components. Type 304L has a lower nitrogen content (0.10-percent maximum) than Type 304LN (0.10 percent to 0.16 percent). Because of its lower nitrogen content, Type 304L is not as strong as Type 304LN and high-carbon SS Types 304 and 304H. Relative to Type 304L, the higher nitrogen content in Type 304LN results in a strengthened material that is both resistant to sensitization (due to the low carbon content) and possesses the higher tensile and yield strength properties of Type 304 and 304H materials. Types 304LN, 304, and 304H all possess an ASME Code minimum tensile strength of 517.1 megapascals (MPa) (75 kilopounds per square inch (ksi)) and a minimum yield strength of 206.8 MPa (30 ksi). Type 304L possesses an ASME Code minimum tensile strength of 475.7 MPa (69 ksi) and a minimum yield strength of 172.4 MPa (25 ksi). Type 304L SS may be used for reactor internal core support structural applications where its lower strength properties are permitted, in accordance with ASME Code, Section III.

According to the applicant, the primary justification for adding Types 304, 304H, and 304L SSs to DCD Section 4.5.2.1 is the application of these materials in currently operating Westinghouse plants. The applicant indicated that SCC has not been experienced in reactor internal core support structure components fabricated with any of these three materials. The use of Type 304L, the lower strength grade, is unconditionally approved for currently operating Westinghouse reactors where its lower strength properties are permitted. The applicant stated that it previously implemented a change from Type 304 to Type 304H for certain reactor internal components in operating plants as the reactor design evolved over time; the applicant provided a list of plants to demonstrate the extensive application of Type 304H SS in reactor internal core support structure components for these later designs. The applicant noted that, for many of the later plants, carbon content in several Type 304 components was limited to a specified range of 0.04 percent to 0.08 percent. This was stated as being the equivalent of a Type 304H SS with a more restrictive 0.08-percent upper limit on carbon content. The applicant stated that, to be consistent with these later plants, the carbon content in Type 304H SSs should be limited to a maximum of 0.08 percent for reactor internal core support structure components in the AP1000 plant. The Westinghouse Utilities Requirements Document requires the use of RG 1.44. Accordingly, all fabricators will be required to establish maximum heat inputs for each welding process with respect to the maximum carbon content for each SS type.

Overall, the staff found that the applicant provided sound justification for the addition of Types 304, 304H, and 304L SSs to the material specifications in DCD Revision 16, Section 4.5.2.1. In particular, the staff noted that the currently operating Westinghouse plants have not experienced problems with IGSCC in reactor internal core support structure components fabricated from these materials. Furthermore, the staff noted that DCD Revision 16, Section 4.5.2.1, specifies that fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with RG 1.44.

In an RAI dated March 11, 2007, the staff requested that the applicant clarify or elaborate on several issues. In RAI-TR31-001, Question 1, Part a, the staff noted an inconsistency in the language in Section 1.2, “Introduction and Brief Description of Change,” of TR-31. Specifically, the staff noted that Section 1.2 of TR-31 states that DCD Revision 15, Section 4.5.2.1, currently specifies Type 304LN SS for reactor internal core support structure components. However, Section 1.2 also states that reactor internal components were designed using Types 304, 304H, and 304L SSs. Therefore, the staff requested that the applicant clarify whether it changed the AP1000 design for the reactor internal components after the issuance of DCD Revision 15 to include these additional SS grades. In its response to RAI-TR31-001, Question 1, Part a, the applicant indicated that it did change the design of the AP1000 reactor internal components after the issuance of Revision 15 of the DCD to include Types 304, 304H, and 304L SSs in addition to Type 304LN SS as potential materials of construction for AP1000 reactor internal components. The staff concludes that this response resolved RAI-TR31-001, Question 1, Part a, because the applicant adequately clarified the statement made in Section 1.2 of TR-31 pertaining to reactor internal component design.

In RAI-TR31-001, Question 1, Part b, the staff requested that the applicant list the materials of construction for each reactor internal component based on the newly proposed reactor internal material specifications identified in TR-31. In its response to RAI-TR31-001, Question 1, Part b, the applicant provided a table depicting the materials of construction for each of the reactor internal components. The staff evaluated this table and determined that the predominate materials of construction for all major core support structures, excluding bolting, are essentially limited to Types 304, 304H, 304L, and 304LN SSs. Furthermore, the staff confirmed the exceptions to the use of the 304-series SSs identified previously, specifically the use of nickel-based Alloy 690 for the clevis inserts, nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts, nickel-based Alloy 750 for the irradiation specimen springs, and the addition of Stellite™ 6 hardfacing for wear surfaces on the radial keys, clevis inserts, and alignment pins. These exceptions were verified to be applicable only to these specific components. The high yield and tensile strength properties and corrosion resistance of nickel-based alloys justify their use for these specific components. These alloys are all acceptable in accordance with ASME Code, Section III. Therefore, the staff identified no safety-related issue associated with their use in these instances. DCD Revision 15, Section 4.5.1.3, previously addressed the qualification of Stellite™ 6 hardfacing for use in RCSs. The staff concludes that the applicant adequately addressed RAI-TR31-001, Question 1, Part b, because it provided a comprehensive list of material specifications for each reactor internal component that is consistent with DCD Revision 16, Section 4.5.2.1.

In RAI-TR31-002, Question 2, the staff requested that the applicant elaborate further on how it addressed the susceptibility of Type 304 and 304H SSs to various forms of corrosion and SCC, where welding on components fabricated using these materials could result in sensitization due to chromium depletion at grain boundaries. In its response to RAI-TR31-002, Question 2, the applicant reiterated its assertion that these SS grades have been used extensively for reactor internal components for currently operating Westinghouse plants. The applicant further stated that the available technical data on environmental degradation applicable to the currently operating Westinghouse plants are also applicable to the design of AP1000 reactor internal components. In addition, these materials have been assessed for a reactor internal component design life of 60 years, with respect to known mechanisms of IASCC and void swelling. DCD Revision 16, Section 4.5.2.1, states that internal material reliability programs are addressing the susceptibility to IASCC and void swelling in reactor internal components identified in the currently operating Westinghouse fleet, and the selection of materials for AP1000 reactor internal components considers information developed by these programs. TR-12, which was provided in WCAP-16620-P, Revision 0, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant," dated July 31, 2006, addresses the evaluation of AP1000 reactor internal components for potential susceptibility to IASCC and void swelling over the 60-year design life and the application of the IASCC and void swelling criteria established by the above material reliability programs. The staff concludes that the applicant's response to RAI-TR31-002, Question 2 was acceptable because it demonstrated that the applicant had adequately addressed the susceptibility of reactor internal components fabricated from Types 304 and 304H SSs to known corrosion and SCC phenomena for reactor internal components.

Based on the above discussion, the staff concludes that the applicant provided an appropriate technical justification for the reactor internal material specification changes proposed in TR-31 because these proposed changes meet the requirements 10 CFR 50.55a and GDC 1 and will not adversely impact the safety of the AP1000 reactor design. Furthermore, the staff concludes that DCD, Section 4.5.2.1, fully represents these material changes. Therefore, the staff concludes that the DCD changes proposed by the applicant in TR-31 are acceptable.

The staff reviewed the proposed changes as they relate to Revision 16 of the AP1000 DCD. The AP1000 DCD incorporated the proposed changes identified in TR-31. Accordingly, these changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD in accordance with 10 CFR 52.63(a)(1)(cii).

4.5.2.2 Conclusion

The staff finds that the changes to the material specifications for the reactor internal components proposed in TR-31 are technically acceptable because these changes meet the requirements of 10 CFR 50.55a and GDC 1 and will not adversely impact the safety of the AP1000 reactor design. Furthermore, the staff finds that the TR-31 conclusions regarding design changes related to the material specifications for AP1000 reactor internal components are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

4.5.3 Changes to In-Core Instrument Guide Tubes

In DCD Revision 17, the applicant proposed a change related to the replacement of in-core instrument guide tubes with QuickLoc assemblies, as well as several editorial modifications to DCD Tier 1, Section 2.1.3, “Reactor System.”

4.5.3.1 Evaluation

The technical change involved the replacement of in-core instrument guide tubes with QuickLoc assemblies, as supported by APP-GW-GLE-016, “Impact of In-core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package (IHP).” Also, this change included specifying the incore instrumentation QuickLoc assemblies as ASME Code Section III Classification and Seismic Category I classification in Tier 1 Table 2.1.3-1, and the QuickLoc assemblies as the pressure boundary components in Table 2.1.3-2. The staff reviewed APP-GW-GLE-016 and issued RAI-SRP15.4.8-SRSB-01 related to the rod ejection analysis in DCD Section 15.4.8. The staff determined the QuickLoc-related changes, as described in DCD Revision 17, Tier 1, Section 2.1.3, to be acceptable.

The first editorial modification is found in Table 2.1.3-2, which includes Design Commitment 2.a, and stipulates inspections, tests, analyses, and acceptance criteria (ITAAC) for the reactor upper internals rod guide arrangement. The applicant made an editorial change to refer to the correct figure describing the reactor upper internals rod guide arrangement. Specifically, Revision 15 listed Figure 2.3.1-1 instead of the correct figure (Figure 2.1.3-1).

The applicant made additional editorial modifications throughout Section 2.1.3 to correct the owrd order of “rod cluster control assemblies” (from “rod control cluster assemblies”) and to clarify that the fuel assemblies are located in the containment location only after fuel loading (located in auxiliary building prior to fuel loading). The staff found these changes to be acceptable.

4.5.3.2 Conclusion

Based on the above evaluation, the staff concludes that the changes to DCD Tier 1, Section 2.1.3, are acceptable.

4.6 Functional Design of Reactivity Control Systems

The reactivity control systems for the AP1000 facility are the control rod drive system (CRDS), the reactor trip system, and the passive core cooling system, which can affect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents.

4.6.2 Evaluation

In DCD Revision 17, Section 4.6, the applicant proposed the following revisions to Section 4.6.1, “Information for Control Rod Drive System”: (1) a clarification that DCD Figure 4.2-8 provides the configuration of the driveline, including the CRDM, not the layout of the CRDS; (2) the deletion of the statement that the CRDM outer shroud is an integral portion of the head lifting system; and (3) the deletion of the “conduits for the in-core instrumentation” from the components located among the CRDM and supported by the integral head package.

The staff reviewed these revisions and concluded that they are editorial in nature. The applicant made the latter two revisions for the purpose of accuracy and consistency with the DCD Section 3.9.4.1.1 modifications to the integrated head package and redesign of the in-core instrumentation. These changes do not alter the functional design of the reactivity control systems and are, therefore, acceptable.

4.6.3 Conclusion

The NRC staff concludes that revisions to AP1000 DCD Section 4.6 are acceptable because the changes are editorial and do not alter the functional design of the reactivity control systems.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

5.1.1 Evaluation

The summary description of the “Reactor Coolant System and Connected Systems,” of the AP1000 reactor coolant system (RCS) and connected systems, as well as their design bases, were evaluated by the staff of the U.S. Nuclear Regulatory Commission (NRC) in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.” Conforming changes to this section have been included in the revisions to the design control document (DCD) to reflect the design and bases changes to the reactor coolant pump (RCP), steam generators (SG), pressurizer, and normal residual heat removal system (RNS) designs that are discussed in Sections 5.4.1, 5.4.2, 5.4.5, and 5.4.7, respectively, as well as other DCD changes addressed in this chapter.

In Section 5.1 of the DCD, the applicant modified Figure 5.1-5, “Reactor Coolant System Piping and Instrumentation Diagram” (Sheet 1 of 3). The applicant stated that Figure 5.1-5 was modified by relocating the Loop 1 narrow-range and diverse actuation system (DAS) resistance temperature detector (RTD) upstream of the pressurizer surge nozzle, and by relocating the wide-range RTD upstream of the passive residual heat removal (PRHR) nozzle.

The applicant explained that the present Loop 1 location of the narrow-range device is a problem because pressurizer outsurges will cause erroneously high signals for some of the T-hot channels. The faster the RCS temperature is decreasing, the larger the resulting outsurge will be, and the higher the indicated Loop 1 temperature. The applicant concluded the most appropriate of several solutions is to relocate the narrow-range device upstream of the pressurizer surge nozzle. The staff agrees that the relocation of these RTDs would alleviate the influence of pressurizer outsurge and finds this to be acceptable.

In addition, AP1000 post-accident monitoring requirements also require that both wide-range T-hot RTDs be located at the top of the hot legs in order to detect voids. DCD Table 7.5-1, “Post-Accident Monitoring System,” specifies two RCS wide-range T-hot measurements and one PRHR heat exchanger inlet temperature measurement. Therefore, the wide-range protection and safety monitoring system (PMS) hot leg RTD (TE-135A) needs to be relocated upstream of the PRHR nozzle to validate PRHR post-accident monitoring requirements. This location would also provide the desired direct post-accident reactor outlet temperature and PRHR inlet temperature. In conjunction with the Loop 2 hot leg wide-range RTD (TE-135B), the relocation of TE-135A satisfies the need for two wide-range T-hot and one PRHR inlet temperature for post-accident monitoring. In addition, the post-accident monitoring requirements also specify that both wide-range T-hot RTDs be located at the top of the hot leg in order to detect voids. Therefore, Note 24 is added to Figure 5.1-5 to specify that the thermal wells for the Loop 1 and Loop 2 wide-range RTDs (TE-135A and TE-135B) be located at the upper half of the hot leg. The staff finds this to be acceptable.

The information above was provided by the applicant in draft format but was not formally submitted to the staff. In a letter dated May 28, 2009, the applicant documented the details related to relocating the narrow-range and wide-range RTDs as discussed above which resolves this issue.

5.1.2 Conclusion

The staff has reviewed the proposed changes to Section 5.1, "Summary Description." Based on the evaluation described above, the staff concludes that the proposed changes to AP1000 DCD Figure 5.1-5 (Sheet 1 of 3) are acceptable pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(a)(1)(vii), "Finality of standard design certifications," on the basis that they contribute to the increased standardization of the certification information.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with Codes and Code Cases

The staff approved Section 5.2.1.1, "Applicable Code Cases," of the AP1000 DCD, Revision 15, in the certified design. The applicant has proposed to make the following changes to Section 5.2.1 of the certified design:

5.2.1.1 Compliance With 10 CFR 50.55a

AP1000 DCD Tier 2, Section 5.2.1.1, "Code Compliance with 10 CFR 50.55a," uses the baseline code of the 1998 Edition throughout and including the 2000 Addenda for evaluations of the safety analysis and the AP1000 design certification (DC), except for 1989 Edition, 1989 Addenda for Articles NB-3200, NB-3600, NC-3600, and ND-3600 related to piping design. In a letter dated January 28, 2009, the applicant proposed changes to AP1000 DCD Tier 2, Section 5.2.1.1 to use 1989 Edition, 1989 Addenda for Articles NB-3210, NB-3620, NB-3650, NC-3620, NC-3650, ND-3620 and ND-3650. These proposed limitations represent only portions of Articles NB-3200, NB-3600, NC-3600, and ND-3600 disallowed by 10 CFR 50.55a(b)(1)(iii), "Codes and standards," for seismic design of piping. In request for additional information (RAI)-SRP5.2.1-EMB-03, the staff requested the applicant confirm whether the AP1000 piping design utilized the 1998 Edition, 2000 Addenda Article NB-3220 for the piping design. The staff also requested the applicant explain how the proposed changes would meet the requirements of 10 CFR 50.55a(b)(1)(iii), including the code requirements relevant to "Reversing Dynamic Loading in Piping," first introduced in the American Society of Mechanical Engineers (ASME) 1994 Addenda.

In the proposed changes to AP1000 Tier 2, DCD Section 5.2.1.1, the applicant indicated that it would use the 1989 Edition and the1989 Addenda sub-articles NB-3620, NC-3620, ND-3620, NB-3650, NC-3650 and ND-3650 for the seismic design of piping. The staff notes that these are the sub-articles describing the alternative provisions for seismic design of piping that were introduced in the 1994 Addenda Sections NB-3600, NC-3600, and ND-3600. The use of the above proposed sub-articles is consistent with the provisions of 10 CFR 50.55a(b)(1)(iii) and is, therefore, acceptable. Regarding the requirements in 10 CFR 50.55a(b)(1)(ii) relating to weld leg dimensions for socket welds, AP1000 piping design will comply with the requirements of 10 CFR 50.55a(b)(1)(ii) for socket weld dimensions, such that DCD Section 5.2.1.1 includes specific requirements including primary stress indices and stress intensification factor consistent with the requirements of 10 CFR 50.55a(b)(1)(ii). In its response dated April 1, 2009, to RAI-SRP5.2.1-EMB-03, the applicant stated that the AP1000 design does not utilize the alternative provisions introduced in NB-3200 from the 1994 Addenda for seismic design of piping. The applicant stated that AP1000 DCD Section 5.2.1.1 will be revised to include the use of the1989 Edition, 1989 Addenda for Subarticle NB-3220. The staff considers the applicant's response and the planned DCD changes to be acceptable. In a subsequent revision to the

AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.2.1.2 Applicable Code Cases

The staff approved Section 5.2.1.2, "Applicable Code Cases," of the AP1000 DCD, Revision 15, in the certified design. The applicant has proposed to make the following changes to Section 5.2.1.2 of the certified design:

1) DCD Tier 2 Section 5.2.1.2 was revised to reference Section 5.2.6.1, which includes a commitment that the combined license (COL) applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code Edition and Addenda as well as Code cases approved subsequent to the DC.

5.2.1.2.1 Evaluation

The staff reviewed this change in accordance with Section 5.2.1.2, "Applicable Code Cases," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." The staff reviewed all changes identified by change marks in the AP1000 DCD. The staff did not re-review descriptions and evaluations in the AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes.

The applicant revised DCD Tier 2, Section 5.2.1.2, to reference Section 5.2.6.1, which includes a commitment that the COL applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code Edition and Addenda as well as Code cases approved subsequent to the DC. To ensure that appropriate Code cases are applied for inspection and examination, in RAI-SRP5.2.1-EMB01, the NRC asked the applicant to provide Code cases similar to those in DCD Table 5.2-3, which are applicable to regulatory guide (RG) 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI Division 1," and RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code."

In a letter dated July 18, 2008, the applicant responded to RAI-SRP5.2.1-EMB-01 and indicated that the Code cases to be used for inservice inspection and testing should be included in the programs developed for these activities. The final safety analysis report (FSAR) in the COL application describes these programs. DCD Section 5.2.1.2 is not an appropriate place to identify the Code cases expected to be used for inservice inspection and inservice testing. However, the COL FSAR incorporates by reference the DCD statement that if Code cases other than those included in DCD Table 5.2-3 are used, a reconciliation review will be performed. It is important that the applicant provide the baseline Code cases similar to those in Table 5.2-3 that will be incorporated by reference in the COL application for use in inspection and testing. Without those baseline Code cases, the staff cannot determine whether the Code cases are the correct revision or if additional new Code cases are needed in the COL application to meet the requirements of 10 CFR 50.55a(a)(3), (b)(5), and (b)(6). The staff identified this as Open Item OI-SRP5.2.1-EMB-01.

In a letter dated July 15, 2009, the applicant stated that DCD Section 3.9.8.4 will be revised to require that the inservice test program identify the ASME Operation and Maintenance (OM) Code cases used. The applicant also indicated that DCD Section 5.2.1.2 would be revised to note that ASME Code, Section XI and the ASME OM Code and associated Code cases are not directly used in the design of the reactor coolant pressure boundary. The ASME Code cases used in the inservice inspection and inservice testing programs would be determined as the COL information based on regulatory guidance and regulations, since RG 1.192 is not applicable to the AP1000 DC. In considering the applicant's planned changes to have Code cases used for the preservice and inservice testing included in the COL information, the staff finds the supplemental information in the letter dated July 15, 2009, to be reasonable and acceptable, based on the requirements in 10 CFR 50.55a. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In DCD Tier 2, Section 5.2.1.2, the applicant stated that the use of any Code case not approved in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," and RG 1.85, "Materials Code Case Acceptability—ASME Section III, Division 1," on Class 1 components is authorized as provided in 10 CFR 50.55a(a)(3) and the requirements of the DC. In RAI-SRP5.2.1-EMB-02, the staff asked the applicant to remove the reference to RG 1.85 from the DCD since the NRC withdrew it in 2003.

In a letter dated June 10, 2008, the applicant responded to RAI-SRP5.2.1-EMB-02 and noted that RG 1.85 is retained in the DCD for historical reasons. It also indicated that because the ASME Code used for the design of the AP1000 is the 1998 Edition, 2000 Addendum, RG 1.85 may include Code cases that are of interest. Since the last revision (Revision 31) of RG 1.85 includes Code cases with dates only up to 1994, the use of RG 1.85 is unrelated to meeting the requirements of the ASME Code, 1998 Edition through 2000 Addenda. Instead, the NRC withdrew RG 1.85 in 2003 but incorporated all Code cases from the original RG 1.85 into RG 1.84 and will continue to do so in its future updates. Therefore, the applicant should revise the DCD Tier 2 to RG 1.84 rather than RG 1.85 while updating the DCD Tier 2 to include the up-to-date information. The staff identified this as Open Item OI-SRP5.2.1-EMB-02.

In a letter dated June 17, 2009, the applicant indicated that DCD Section 5.2.1.2 would be revised to remove the references to RG 1.85. The applicant also indicated that the discussion of conformance to RG 1.84 in Appendix 1A would be revised from Revision 31 to Revision 32 to be consistent with the information in Table 1.9-1. The staff finds the applicant's response and the planned changes to the AP1000 DCD to be acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Revision 17 of AP1000 DCD included three new Code cases used in the AP1000 design, which were added to Table 5.2-3. These Code cases are N-655, "Use of SA-738, Grade B, for Metal Containment Vessels, Class MC, Section III, Division 1"; N-757, "Alternative Rules for Acceptability for Class 2 and 3 Valves, NPS 1 (DN25) and Smaller with Welded and Non-welded End Connections other than Flanges, Section III, Division 1"; and N-759-1, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1." In RAI-SRP5.2.1-EMB-04, the staff requested the applicant confirm whether these Code cases were the most recent version and whether they were approved by the NRC in RG 1.84. If not, the staff requested that the applicant provide justification for using these Code cases in the AP1000 design in accordance with the requirements of ASME Code, Section III, NCA-1140.

In its response dated April 1, 2009, to RAI-SRP5.2.1-EMB-04, as supplemented by a letter dated June 8, 2009, the applicant stated that ASME Section III Code Cases N-757 and N-759-1 are not included in Revision 34 of RG 1.84. The staff notes that Code Case N-655 was conditionally approved by the NRC in Revision 33 of RG 1.84. The applicant also indicated that the AP1000 design would apply the latest Code Cases N-655-1, N-757-1 and N-759-2 as identified in its letter dated June 8, 2009. The applicant provided justification required by 10 CFR 50.55a for the use of these Code cases as discussed in the following paragraphs.

See Section 3.8.2.5 of this report for the evaluation of ASME Code Case N-655. Since Code Case N-655-1 revised Code Case N-655 by replacing Supplementary Requirement S-17 with Supplementary Requirement S-1, and is considered to be equivalent with respect to controlling the quality of the SA-738 material, the staff finds that Code Case N-655-1 provides an acceptable level of quality and safety and is authorized pursuant to 10 CFR 50.55a(a)(3)(i). The applicant indicated that DCD Table 5.2-3 would be revised to include Code Case N-655-1 in lieu of N-655. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Code Case N-757 allows the use of alternate rules for the design of instrument, control and sampling line valves, Class 2 and 3 NPS 1 (DN25) and smaller, with welded and non-welded end connections other than flanges. The ASME Code requirements for the design of these valves are in Articles NC-3500 and ND-3500 for ASME Code Class 2 and 3 valves. The standard design rules in Paragraphs NC-3512 and ND-3512 require that the minimum wall thickness satisfy the thickness requirements specified in the valve standard ASME B16.34, “Valves- Flanged, Threaded, and Welding End.” Paragraphs NC-3513 and ND-3513 provide the alternate design rules that may be used in place of NC-3512 and ND-3512 when permitted by the design specification. However, these alternate rules only apply to valves with butt welding end connections and socket welding end connections. Code Case N-757-1 allows the use of the alternate design rules for welded and non-welded end connections other than flanges, in the design of small valves. Code Case N-757-1 states that these valves may meet the design requirements of Section III, Division 1, Class 2 and 3 rules in Paragraphs NC-3512 and ND-3512 provided specific additional requirements are met. Based on a discussion in a conference call with the staff on June 19, 2009, for the use of Code Case N-757-1, the applicant addressed, in a letter dated July 2, 2009, issues regarding the operating experience of non-welded valves identified in NRC Information Notice (IN) 84-55, “Seal Table Leaks at PWRs,” and IN 92-15, “Failure of Primary System Compression Fitting.” The applicant stated that information and cautions would be provided in design documents (e.g., design specifications and instruction manuals) including: (a) not mixing the parts from one manufacturer to another; (b) following manufacturer's recommended instructions for installing compression fittings; and (c) providing training and procedures for personnel performing the work. The staff concludes that the use of Code Case N-757-1 will provide an acceptable level of quality and safety for the design of instrument, control and sampling line valves, Class 2 and 3 NPS 1 (DN25) and smaller, with welded and non-welded end connections other than flanges, and is, therefore, authorized pursuant to 10 CFR 50.55a(a)(3)(i). The applicant indicated that DCD Table 5.2-3 would be revised to include Code Case N-757-1 in lieu of N-757. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Code Case N-759-2, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1," is intended for use in the design of AP1000 to address an issue with the primary side depressurization transients. Code Case N-759-2 provides an alternative methodology for the SG tube collapse analysis based on theoretical buckling equations and buckling tests on fabricated cylindrical tubes.

In a letter dated July 2, 2009, the applicant provided supplemental information stating that the use of theoretical buckling equations and buckling tests provides assurance that Code Case N-759-2 would provide an acceptable level of quality and safety. The applicant also stated that compliance with the existing rules of the ASME Code without the use of Code Case N-759-2 would require a re-design of the AP1000 SG tube bundle. The staff finds that the use of Code Case N-759-2 provides an acceptable level of quality and safety and is, thus, authorized pursuant to 10CFR 50.55a(a)(3)(i). The applicant indicated that AP1000 DCD Tier 2, Table 5.2-3 would be revised to replace Code Case N-759-1 with N-759-2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Revision 17 to the AP1000 DCD, Introduction, Table 1-1, “Index of AP1000 Tier 2 Information Requiring NRC Approval for Change,” includes Tier 2* information requiring NRC approval for change. AP1000 DCD Tier 2, Table 5.2-3 also includes Tier 2* information as specified in the footnotes to the table. In RAI-SRP5.2.1-EMB-05, the staff requested the applicant revise Table 1-1 to be consistent with Table 5.2-3. In its response to RAI-SRP5.2.1-EMB-05, the applicant indicated that it had identified inconsistencies and an omission in Table 1-1 and planned to make several changes. For example, the reference to a specific Code case would be expanded to cover all ASME Code cases with a reference to Table 5.2-3. A note specifying that the 2001 Edition of the ASME Code, Section III, including 2002 Addenda, applies to containment design would be included in Table 1-1. The table would also be revised to remove incorrect references to Sections 3.8.2.5 and 5.2.1.1 for this item. An item would be added to Table 1-1 listing the baseline ASME Code Edition and Addenda with reference to Section 5.2.1.1. The staff considers the applicant’s planned changes to the DCD provide consistency in the applicable DCD sections and are, therefore, acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

By letters dated April 28 and June 8, 2009, the applicant requested the use of Code Case N-782, “Use of Code Editions, Addenda, and Cases Section III, Division 1,” for the AP1000 design. This code case is not included in RG 1.84, Revision 34. As required by 10 CFR 50.55a(a)(3), the applicant requested approval for the use of this ASME Code case as a proposed alternative to the rules of 10 CFR 50.55a.

Code Case N-782 provides that the Code Edition and Addenda endorsed in a design certified by the regulatory authority may be used for systems and components constructed to ASME Code, Section III requirements. This Code case updates Paragraph NCA-1140 of ASME Code, Section III to address the licensing process under 10 CFR Part 52, “License, certifications, and approvals for nuclear power plants.” The applicant indicated that the use and approval of Code Case N-782 was needed to align ASME Code requirements to the ASME Code Edition and Addenda cited in AP1000 DCD Tier 2, Section 5.2.1.1. A reference to Code Case N-782 would be included in component and system design specifications and design reports to permit certification of these specifications and reports to the Code Edition and Addenda cited in the DCD and approved by the NRC. The applicant indicated that the use of Code Case N-782 facilitates the use of the ASME Code Edition and Addenda included in the AP1000 DC. Therefore, it would provide the same level of quality and safety as was included in the information reviewed for the AP1000 DC. The information provided in the applicant’s letter is generic and applies to all COL applicants referencing the AP1000 DC. The staff concludes that the use of Code Case N-782 provides an acceptable level of quality and safety and is, therefore, authorized pursuant to 10 CFR 50.55a(a)(3)(i). AP1000 DCD Tier 2, Table 5.2-3 will be revised to reference Code Case N-782. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.2.1.2.2 Conclusions

Based on the information provided in the applicant's responses to the RAIs, the staff finds that the AP1000 DCD amendment conforms to the guidance provided in RG 1.206, "Combined License Applications for Nuclear Power Plants." This satisfies the requirements of 10 CFR 50.55a and General Design Criteria (GDC) 1, "Quality Standards and Records," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities," and, therefore, is acceptable.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Evaluation of Zinc Addition to the Reactor Coolant

5.2.3.1.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant proposed design changes to incorporate the ability to inject zinc into AP1000 reactor coolant. In a letter dated April 5, 2006, the applicant submitted TR-32, "Zinc Addition," APP-GW-GLN-002, to provide technical justification for the proposed zinc addition into the reactor coolant.

The request consists of a modification to the AP1000 design to incorporate the ability to inject a small quantity of zinc acetate into the RCS. Operation with chemical zinc in the cooling system has been shown to change the oxide film on primary components, reducing occupational radiation exposure and the potential for crud formation, crud deposition on the fuel rods, and subsequent power shifts. The applicant proposed to provide zinc addition as an optional mode of operation.

Implementation of the requested zinc injection capability in the AP1000 results in the following Tier 2 changes to the DCD:

1. Add note 7 in Table 5.2-2, "Reactor Coolant Water Chemistry Specifications," to specify the maximum zinc concentration.

2. In Table 6.2.3-1 revise the chemical and volume control system hydrogen injection to the RCS line isolation device to be normally open.

3. Revise the third paragraph of Section 9.3.6.2.1, "Purification," to the effect that the mixed bed demineralizers will also remove zinc.

4. Add Section 9.3.6.2.3.3, "Zinc Addition," to the effect that zinc may be added.

5. Revise Figure 9.3.6-1 to include: Valve V092 is changed to normally open, a reducer is added downstream of V065, the portion of the H2/ZINC ADD line from the reducer to the return line is renumbered as L064, and the specification of L064 is changed to .5" BBC.

5.2.3.1.2 Evaluation

GDC 4, "Environmental and Missile Dynamic Effects Design Bases," requires that structures, systems, and components (SSCs) important to safety shall be appropriately protected against environmental and dynamic effects. The staff reviewed changes related to this section to ensure the compatibility of components with the environmental conditions created by the addition of zinc.

In letters dated August 1, 2006, and September 28, 2006, the staff requested additional information regarding the effects of zinc injection. The applicant responded in letters dated September 8, 2006, and December 12, 2006. The staff reviewed the responses and included that information in the overall evaluation, which follows.

Zinc acetate will be added using the same piping and valving as the hydrogen addition. The proposed hardware change is to replace a portion of the 2.5 centimeter (cm) (1-inch (in)) pipe (downstream from the containment isolation valve) with a heavier wall 1.3 cm (½-in) pipe. This will reduce the piping volume and substantially reduce the transit time for the hydrogen and the zinc acetate injected material. The integrity of the pressure boundary is not affected. Both hydrogen and zinc acetate injections have low transit velocities; thus, flow stability is not a problem. The containment isolation function signal, the containment isolation, the valve designation as active (Table 3.9-12 in the DCD), the safety-related mission, the inservice testing type and frequency requirements (Table 3.9-16), and the valve functional requirements for containment isolation (Tier 1, Table 2.3.2-1) are not affected.

The staff concludes that zinc addition in the primary coolant reduces radiation fields and the formation of crud, which may result in increased personnel exposure and in axial power shifts, respectively. Regarding the effect of zinc in reducing primary water stress-corrosion cracking, the applicant clarified that it no longer takes credit for the mitigation of this type of cracking based on zinc addition. The applicant believes that there is sufficient margin in the selection of new materials and that credit from zinc addition is not needed. The staff finds that the presence of zinc in the primary water will not cause aging-related degradation; therefore, it is acceptable.

The staff questioned the effect of the thin oxide film that forms in the presence of zinc with respect to the potential increase of the heat transfer coefficient and the potential increase in fuel and cladding operating temperature. The applicant submitted additional information in the form of cladding oxidation results that demonstrated that existing plant operational data (using ZIRLO™ cladding) did not exhibit increased oxidation, suggesting that the cladding is operating at comparable temperatures. The staff finds that the presence of zinc meets the requirements of GDC 4 and does not decrease the cladding-to-coolant heat transfer coefficient; thus, the staff finds the presence of zinc in the primary water to be acceptable. These changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant.

5.2.3.1.3 Conclusions

On the basis of its review of TR-32, the staff finds that the requested modification for zinc addition to the primary water meets the requirements of GDC 4 and is acceptable for ZIRLO™ fuel cladding. Furthermore, the staff finds that the TR-32 conclusions regarding the evaluation for zinc addition to the primary water are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.2.3.2 Evaluation of Reactor Coolant Pressure Boundary Materials

5.2.3.2.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant proposed design changes related to the AP1000 pressure boundary materials. In a letter dated May 24, 2007, the applicant submitted TR-33, “Pressure Boundary Material Change” APP-GW-GLN-009, Revision 1, which provides technical justification for the design changes related to the AP1000 pressure boundary materials. The pressure boundary materials selection appears in DCD Tier 2, Section 5.2, Table 5.2-1. The pressure boundary materials changes identified in TR-33 are: (1) the revision of some material designators to be consistent with the ASME Code of Record for the RCS pressure boundary components, which is the 1998 Code and 2000 Addenda; (2) the correction of discrepancies in the AP1000 DCD in the specification of materials for some components; (3) the addition of materials to DCD Table 5.2-1 to address previously identified issues, to provide fabrication flexibility, and to ensure adequate material supply; (4) the relaxation of the maximum copper limit allowed in the reactor vessel beltline forging and weld material to reduce schedule risk and cost while maintaining the required performance; and (5) the relaxation of the maximum delta ferrite limit in weld materials to reduce schedule risk and cost while maintaining the performance requirements.

Revision 17 revised Section 5.2.3.1 of the AP1000 DCD, Tier 2, to include ASME Code Filler Metal Specifications SFA 5.1 and 5.17 for carbon steel, ASME Code Filler Metal Specification SFA 5.22 for stainless steel which allows the use of fluxed-cored filler metal to be used for welding the root pass, and ASME Code Filler Metal Specification SFA 5.30 for consumable inserts. Revision 17 also revised AP1000 DCD, Table 5.2-1, to allow an option to use carbon steel (SA-508, Class 1A) instead of alloy steel (SA-508, Grade 3, Class 2) on the pressure forgings (including nozzles and tubesheets for the SG), and allows an option to use carbon steel (SA-508, Grade 1) instead of stainless steel (SA-336, Grades F304, F304L, F304LN, F316, F316L and F316LN) on the pressure forgings for the RCP. In addition, Table 5.2-1 of the DCD was changed in Revision 17 to add material specification SA-338 for the pressurizer nozzle safe-ends, and to replace material specifications SA-312 and SA-376 for seamless pipe with material specification SA-479 for hot-rolled or cold-rolled bar stock with a unified numbering system (UNS) designation S21800.

The staff reviewed information in the AP1000 TR-33 that supports Revision 16 to the AP1000 DCD, along with the associated changes made in Revision 17. The staff’s findings are summarized below.

5.2.3.2.2 Evaluation of Revision 16 Changes

This evaluation addresses the impact of the changes identified in TR-33 related to the Class 1 pressure boundary materials, in such areas as the reactor vessel and internals, pressurizer, and SGs.

a. ASME Code of Record Update

Reactor vessel components (head plates, shell courses, shell flange, and appurtenances to the control rod drive mechanism (CRDM) material designators) were revised to be consistent with the ASME Code of Record for the RCS pressure boundary components, which is the 1998 Code and 2000 Addenda. These changes have no impact on the safety evaluation performed by the staff because the materials identified continue to meet the requirements of the ASME Code, Section III. Hence, the changes are acceptable.

b. Material Addition

Reactor vessel components (appurtenances to the CRDM, instrumentation tube appurtenances, and monitor tubes) materials have been added. The added materials have been used in pressurized-water reactors (PWRs) in the past. In addition, reactor vessel components (nozzle safe ends, appurtenances to the CRDM, instrumentation tube appurtenances, upper head, monitor tubes, and vent pipe materials) currently identified in the DCD and TR-33 include corresponding class, grade, or type. The identified added materials and the additional information on grades, types, or classes meet the requirements of the ASME Code, Sections II and III in accordance with the guidance of NUREG-0800 Section 5.2.3. Therefore, the inclusion of the identified materials and the additional information on grades, types, or classes presented in TR-33 have no impact on the conclusions reached by the staff in its review of the AP1000 DCD, Revision 15. Hence, the additions are acceptable. Section 4.5.1 of this report discusses the evaluation of the material changes for the CRDM components.

c. Reactor Vessel Beltline Forging and Weld Chemical Composition

The current AP1000 DCD, Revision 15, specifies maximum 0.03 weight-percent (wt%) copper (Cu) for reactor vessel beltline forgings and welds. The copper limits were established to address irradiation embrittlement concerns. TR-33 proposed that the copper limit for the reactor vessel beltline forging and weld material be 0.06 wt%. Forgings with specified maximum copper limits of 0.03 wt% are outside the current practice for the potential forging suppliers since most specifications are commonly in the range of 0.05 to 0.1 wt%.

Changing the wt% of copper impacts the end-of-life (EOL) reference temperature-pressurized thermal shock (RT_{PTS}), which is calculated according to the methodology prescribed in 10 CFR 50.61(c)(1), “Fracture toughness requirements for protection against pressurized thermal shock events.” However, even though the RT_{PTS} value is increased, there is still substantial margin from exceeding the screening criteria limit set forth in 10 CFR 50.61(b)(2). Table 5-1 summarizes the impact of increased copper on the EOL RT_{PTS} (using the current RT_{PTS} criteria).

Table 5-1. Impact of Increased Copper on the EOL RT_{PTS}

	EOL RT_{PTS} CU =.03 wt% Current Limit	**EOL RT_{PTS} CU = 0.06 wt% Proposed Limit**	**EOL RT_{PTS} Screening Criteria**
Beltline Forging	18.9 °Celsius (C) (66 °Fahrenheit (F))	34.4 °C (94 °F)	<132.2 °C (270 °F)
Beltline weld	36.7 °C (98 °F)	64.4 °C (148 °F)	< 148.9 °C (300 °F)

Based on the data shown in Table 5-1, the impact of higher copper content on the beltline weld and beltline forging EOL RT_{PTS} is small and does not challenge the screening criteria of 10 CFR 50.61(b)(2).

Adjusted reference temperature (ART) will be slightly increased because of the higher copper content and that will decrease the allowable operating temperature. However, this will not result in any significant restrictions on plant operations. The pressure-temperature (P/T) curve changes will not significantly affect the low-temperature overpressure protection (LTOP) system evaluation and the resulting parameters for the normal RNS relief valve.

Thus, the impacts of the increase in copper content of the beltline weld and beltline forging are insignificant. Taking into consideration the increase in copper content, the staff's review finds that the requirements of GDC 1 and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for the beltline forgings and weld materials. The materials also meet the requirements of Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 and 10 CFR 50.61. Therefore, the staff's review finds the changes to the reactor vessel beltline forging and weld chemical compositions acceptable.

d. Delta Ferrite Limits

AP1000 DCD Revision 15, specifies an upper delta ferrite limit of 13 ferrite number (FN). The proposed change is to increase the upper-shelf limit to 20 FN and to clarify the acceptable methods to verify the delta ferrite content. The proposed change to the maximum delta ferrite content is 20 FN for filler metal compositions with low molybdenum contents and 16 FN for weld filler materials with higher molybdenum content, such as Types 316/316L. The increase in maximum allowable delta ferrite levels will increase the availability of suppliers and flexibility in fabrication, resulting in a decrease in cost and fabrication time without compromising material performance.

The upper-shelf limit of 20 FN is still within the guidelines established by the staff under RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," which states that weld pad test results showing an average FN from 5 to 20 indicate that the filler metal is acceptable for production welding of Class 1 and 2 austenitic stainless steel components and core support structures. In addition, the upper- and lower-shelf limit FN content is within the acceptance standards under ASME Code, Section III, NB-2000. Based on the discussion above, the change is within the limits prescribed by the staff and the ASME Code and is acceptable.

e. Primary and Auxiliary Piping

Section 3B.2, "Potential Failure Mechanisms for AP1000 Piping," of Appendix 3B, "Leak-Before-Break Evaluation of the AP1000 Piping," to the AP1000 DCD, Revision 15, states that SA312TP316LN- and SA304TP304L-grade steels selected for primary and auxiliary piping are both resistant to intergranular stress-corrosion cracking and wall thinning due to erosion-corrosion effects. The proposed change replaces the references from specific materials to a more generic statement, which calls for Series 300 stainless steel materials. The proposed change also states that these materials were chosen because of their proven operating experience in low- or no-oxygen environments with no incidents for a number of years. The proposed change also states that RG 1.44, "Control of the Use of Sensitized Stainless Steel," "will be used to maintain experiences of the PWR applications for the use of Series 300 stainless steel materials." The applicant also revised Table 5.2-1 of DCD Section 5.2 to include the specific material Types 304, 304L, 316, and 316L, in addition to the current Types 304LN and 316LN used for the reactor coolant pressure boundary components, including piping.

In proposing this change, higher carbon stainless steel materials (Types 304 and 316) could be used in the reactor coolant pressure boundary, including piping, and may be more susceptible to sensitization due to heat treatments or welding. However, for the applications proposed, the reactor coolant environment to which these materials will be exposed is either a low- or no-oxygen environment in part because of the use of oxygen scavenger chemicals such as gaseous hydrogen and hydrazine. Past operating experience with the use of Series 300 stainless steels in such low- or no-oxygen environments has not shown any significant challenges to piping integrity for a number of years. Therefore, the staff concurs that the use of Series 300 stainless steels is the most proven application in the fleet, taking into consideration operational experience. In addition, the DCD mentions RG 1.44 as the guideline followed in the selection and use of the material and the control methods to avoid sensitization. Therefore, the staff finds that the use of these stainless steel materials exposed to PWR reactor coolant water is acceptable. However, the staff notes that the use of these materials on any piping that implements leak-before-break will be verified as part of inspections, tests, analyses, and acceptance criteria (ITAAC) during the COL phase using the actual material properties and final, as-built piping analysis to ensure that the piping using these materials still meets the leak-before-break bounding analysis curves (which were originally evaluated for Types 304LN and 316LN piping material).

f. Steam Generators

TR-33 revises the material designations for pressure boundary components to be consistent with the ASME Code of Record (1998 Code with 2000 Addenda).

Table 5.1-2 of the DCD updates material class and grade designators. For example, the SG tubing material designator was revised from SB-163 TP 690TT to SB-163 N06690. ASME Specification SB-163 identifies the chemical and physical properties of the material and notes that the material is annealed. Section 5.2.3.1.1 of this report states that the SG tubes are made of thermally treated Alloy 690 material. Section 5.2.3.1, “Material Specifications,” of the DCD Revision 15 also states that the SG tubes use Alloy 690 in the thermally treated form.

The material designators for the SG manway closure studs and nuts were also updated to reflect the more correct way to identify bolting material. The studs are SA-193 Gr. B-7 and the nuts are SA-194 Gr. 2H.

The staff finds that the revisions and updates to the DCD related to the SG components do not alter the staff’s conclusions in NUREG-1793 and are consistent with NUREG-0800 Section 5.2.3. Furthermore, the requirements of GDC 1, GDC 30, “Quality of Reactor Coolant Pressure Boundary,” and 10 CFR 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of ASME Code, Section III. Therefore, the staff finds these changes to the DCD related to SG components acceptable.

5.2.3.2.3 Evaluation of Revision 17 Changes

a. Use of Carbon Steel

Revision 17 to the AP1000 DCD proposed the use of carbon steel as an option for reactor coolant pressure boundary components, which include the SG and the RCP. Therefore, the staff requested the applicant provide justification and operating experience on the use carbon steel or remove the use of carbon steel in reactor coolant pressure boundary components.

In its December 31, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 1, the applicant stated that carbon steel is used only for those pressure boundary components which are not normally exposed to the reactor coolant (i.e., RCP components - stator main flange, stator shell and external heat exchanger supports). These components use carbon steel in lieu of 304 stainless steel due to the stresses induced by the studs and the main flange. Specifically, a 304 stainless steel main flange may experience an increase in the main flange stud load at operating conditions due to the thermal expansion between the steel studs and the 304 stainless steel main flange. Therefore, the carbon steel will provide a higher margin against stud/nut to main flange bearing stresses. In addition, Note 4 is proposed to be added to Table 5.2.3-1 of the AP1000 DCD to state the use of carbon steel base material and weld material are limited to only these components of the RCP. In addition, in its April 7, 2010 response to RAI-SRP5.2.3-CIB1-01, Revision 2, the applicant stated that carbon steel will not be used for the SG reactor coolant pressure boundary forgings, and proposed to remove the carbon steel material from Table 5.2-1. The staff notes that carbon steel is only used in 3 components for the RCP that is not in contact with the reactor coolant. Therefore, the staff finds the use of carbon steel for only non-wetted components of the RCP acceptable, and that there is no other use of carbon steel for the reactor coolant pressure boundary. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

b. Use of Flux-cored Welding in Root Pass

Revision 17 to the AP1000 DCD included the use of ASME Code Filler Metal Specification SFA 5.22, which allows the use of fluxed-cored filler metals to be used for welding the root pass in all stainless steel reactor coolant pressure boundary components. The staff has concerns that the use of flux-cored filler metal in the root pass may introduce slag inclusions in the root weld layer in contact with the reactor coolant, thereby providing a crack initiation site. In its August 12, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 0, the applicant stated that the use of flux bearing weld processes for root welds in which the root layer is exposed to the reactor coolant are prohibited unless the backside is first back-gouged to remove the root layer and re-welded from the backside. In addition, in its April 7, 2010 response to RAI-SRP5.2.3-CIB1-01, Revision 2, the applicant proposed Notes 3 and 8 to Table 5.2-1 of the AP1000 DCD, which specifies this prohibition of using flux-bearing filler metal for weld root passes. The NRC staff finds the use of fluxed-cored filler metals with the addition of Notes 3 and 8 to Table 5.2-1 of the AP1000 DCD acceptable since it prohibits the use of flux containing weld processes on the root layer, which would be in contact with the reactor coolant, thereby ensuring the weld integrity by minimizing the occurrence of a crack initiation site in the root layer. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

c. Use of Consumable Inserts

Section 5.2.3.1 of the AP1000 DCD, Revision 17 proposed the use of ASME Code Filler Metal Specification SFA 5.30 for consumable inserts for stainless steel. In addition, Table 5.2-1 of the AP1000 DCD, Revision 17 included Alloy 690 safe ends for the SG nozzle. However, Revision 17 did not include ASME Code Filler Metal Specification SFA 5.30 for consumable inserts joining the SG Alloy 690 safe ends to the RCP casing to ensure full penetration of the weld joint. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, dated July 9, 2010, the applicant stated that welding of the SG Alloy 690 safe end to the RCP casing is performed from both sides to achieve full penetration. The weld design features a double-sided weld since there is access to the second side for back-gouging and re-welding. Therefore, the staff finds the addition of SFA 5.30 acceptable for providing a weld joint to achieve full penetration, particularly for installation closure welds, and the use of a double-sided weld for the SG to the reactor coolant weld to ensure full penetration, which is in accordance with Section III of the ASME Code.

d. RCP – Use of Carbon Steel and the RCP Flywheel Analysis

Since carbon steel will be utilized in the RCP, the staff reviewed how this would affect the flywheel analysis described in Section 5.4.1.4 of this report. In its August 12, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 0, and its December 31, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 1, the applicant stated that carbon steel is only used for the stator main flange, stator shell and the external heat exchanger supports. The staff determined that since the flywheel analysis described in Section 5.4.1.4 only credits the RCP casing, thermal barrier, stator closure ring and the stator lower flange for containing the flywheel, the use of carbon steel has no impact on the flywheel analysis.

However, to clarify the use of the carbon steel base material and welding filler metal for the RCP, the applicant stated in response to RAI-SRP5.2.3-CIB1-01, Revision 1, that full penetration weld joints are used for welding the stator shell (carbon steel) to the stator main flange (carbon steel) and the stator lower flange (stainless steel). The stator shell (carbon steel) to the stator lower flange (stainless steel) is a dissimilar metal weld. The stator shell is buttered with ASME Code Filler Metal Specification 5.9, Classification ER309, and then the weld joining the lower flange to the ER309 buttering uses ASME Code Filler Metal Specification 5.9, Classification 308. Both welds have radiographic inspection requirements pursuant to Section III of the ASME Code, and are surface examined on both the inside and outside diameter surfaces. The staff finds the welding and inspections for the carbon steel components for the RCP will be performed in accordance with the criteria of Section III of the ASME Code for full penetration to ensure the integrity of the welded components.

However, the NRC staff noted that since Filler Metal Specification SFA 5.9, Classification ER309 is used for the weld buttering on the RCP stator shell, which is subsequently post-weld heat treated, the ER309 buttering may become sensitized when exposed to the post-weld heat treatment. In its response to RAI-SRP5.2.3-CIB1-01, Revision 2, dated April 7, 2010, the applicant proposed to add Note 5 to Table 5.2-1 of the AP1000 DCD to clarify the maximum carbon content for stainless steel welds, which would be exposed to post-weld heat treatment that may cause sensitization. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, dated July 9, 2010, the applicant further clarified the carbon content of stainless steels, which meets the guidance in RG 1.44. Therefore, since the maximum carbon content is specified to prevent sensitization and the guidance of RG 1.44 is used when post-weld heat treating stainless steel, the staff finds Note 5 acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

e. Use of EC and EQ

In addition, the applicant's letter dated April 7, 2010, proposed new changes to Table 5.2-1 of the AP1000 DCD to add strip electrodes (EQ) for SFA 5.9 filler materials, and composite cored/stranded electrodes (EC) for SFA 5.9 and SFA 5.14 filler materials from ASME Code, Section II. EQ filler materials currently listed in SFA 5.9 and SFA 5.14 of ASME Code, Section II are used for cladding purposes, and have been used in current operating plants. The use of this filler material allows increase weld surface area deposition, which is particularly beneficial when performing weld cladding of vessels. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, the applicant provided additional information on the use of EC filler material and clarified the use of EQ and EC in Table 5.2-1 of the AP1000 DCD. The applicant stated that EC filler material can be in the form of stranded wire and metal cored wire. The stranded wire is used primarily for automatic gas-tungsten arc welding since it is more flexible, which aids in smooth and consistent wire feeding into the weld puddle. The staff notes that EC filler material provides a more consistent and repeatable weld, which aids in minimizing welding defects, thereby improving the integrity of the weld. Therefore, the staff considers the use of EC filler materials specified in the Section II of the ASME Code acceptable.

The applicant also stated in its response to RAI-SRP5.2.3-CIB1-01, Revision 3, that the metal cored composite filler metal, EC, is used with the gas metal arc welding (GMAW) process, and due to the higher current density of these filler materials, results in higher weld deposition, and can be better controlled by the welder, thereby enabling acceptable welds to be produced consistently. The EC filler material contains no flux, and the mechanical and chemical properties of the resulting weld are identical to other filler materials used for the GMAW process. Therefore, the staff finds the use of EC filler material specified in SFA 5.9 of the ASME Code, Section II, acceptable since these filler materials produce welds that have the same chemical and mechanical properties as other SFA 5.9 filler materials allowed by the ASME Code. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

f. Use of 52M

In a letter dated April 7, 2010, the applicant proposed new changes to Section 5.2.3.1 and Table 5.2-1, Note 7, to include the use of ASME Code Filler Metal Specification 5.14, Type ENiCrFe-7A (UNS N06054) SFA 5.14. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, the applicant clarified that the use of the nickel alloy weld material in SFA 5.14, which includes UNS N06054 (Alloy 52M) would improve weldability. The NRC staff notes that N06054 filler material is more resistant to ductility dip cracking than UNS N06052 and W86152 filler materials, thereby ensuring the integrity of the weld. In addition, other filler material in this classification have been, and are currently being developed to further improve the weldability by minimizing the potential for both hot cracking and ductility dip cracking (microfissuring). Therefore, the staff agrees that the use of these filler material alloys developed for improved weldability for SFA 5.14, which are included in ASME Code, Section II, can be used since these filler materials provide an acceptable level of quality to ensure the integrity of the weld. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

g. Pressurizer Safe-ends

Material Specification SA-338, Grades F316, F316L and F316LN was added in Revision 17 of AP1000 DCD, Table 5.2-1 for the pressurizer safe-ends. However, the NRC notes that this material specification does not exist in ASME Code, Section II. In its response to RAI-SRP5.2.3-CIB1-02, dated December 11, 2009, the applicant changed the material specification to SA-336 to correct the editorial error. The staff finds the addition of Material Specification SA-336 for stainless steel acceptable for use on the pressurizer safe-ends since it is in Section II of the ASME Code and has acceptable operating experience in current plants. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

h. Use of Quickloc and SA-479

Table 5.2-1 of the AP1000 DCD was changed in Revision 17 to replace Material Specifications SA-312 and SA-376 (seamless pipe) with Material Specification SA-479 for hot-rolled or cold-rolled bar stock with no supporting justification or operating experience. In addition, the UNS designation S21800 for Material Specification SA-479 has higher carbon content than Types 304 and 316, along with lower chromium content with no molybdenum additions, thereby making the material susceptible to sensitization if welded. Therefore, the staff requested a discussion on the compatibility of this material with the reactor coolant, along with operating experience. In its response to RAI-SRP5.2.3-CIB1-01, Revision 1, the applicant stated that Material Specification SA-479 replaced SA-312 and SA-376 because the AP1000 design incorporated a Quickloc mechanism to replace the previous incore instrument thimble assemblies (IITAs). The new design includes eight Quickloc penetrations in lieu of the 42 individual IITA penetrations on the reactor vessel head. The applicant also provided operating experience for this material, which includes:

- Used for Quicklocs on several operating plants (Waterford 3, St. Lucie 1 and 2, and Calvert Cliffs 1 and 2) starting from the mid 1990’s.
- Used for core exit nozzle assembly pressure boundary parts since 1988 in approximately 50 plants worldwide.
- Used for various reactor vessel internal parts.

The Material Specification SA 479, UNS 21800 (also known as Nitronic 60) is procured in the solution annealed condition, and is used primarily for areas to minimize wear, galling and fretting due to the additions of silicon and manganese. The staff notes that although Nitronic 60 has a maximum carbon content of 0.10 percent versus 0.08 percent for Types 304 and 316 stainless steels, sensitization of the Nitronic 60 is not a concern for the Quickloc since it is provided in the solution annealed condition and is not subject to welding. In addition, intergranular corrosion tests are performed in accordance with the American Society for Testing and Materials (ASTM) A262-02a, Practice E, as outlined in RG 1.44, to ensure the material is not sensitized. The staff finds that the Nitronic 60 material has satisfactory operating experience in contact with the reactor coolant since the mid 1980’s and, therefore, finds the use of SA 479, UNS 21800 (Nitronic 60) to be acceptable for use in the Quickloc design, and is fabricated to ensure it is not sensitized. However, since the applicant stated that this material is not welded, the staff requested how the Quickloc mechanism is attached to the reactor vessel head.

In its response to RAI-SRP5.2.3-CIB1-03, Revision 0, dated January 15, 2010, and in response to RAI-SRP5.2.3-CIB1-03, Revision 1, dated March 21, 2010, the applicant stated that the Nitronic 60 material is used for the Quickloc plug and Quickloc nut, which are not welded, since these are mechanical parts. The Quickloc plug is inserted into the Quickloc instrument nozzle. The Quickloc instrument nozzles are welded to a weld-buildup of alloy steel on the reactor vessel head. The staff notes that this weld buildup (Material Specifications SFA 5.5, 5.23 or 5.28 of Section II to the ASME Code) is deposited on the top of the reactor vessel head and then machined to form a nozzle that penetrates the reactor vessel head. This weld build up nozzle has corrosion-resistant cladding applied to the inside diameter similar to the rest of the reactor vessel head. The applicant also stated that the buttering (Alloy 52/152) is applied to the top of the weld buildup to facilitate the welding of the Quickloc instrument nozzle to this buttering using Alloy 52/152. Therefore, there is a dissimilar metal weld using Alloy 52/152 attaching the

SA-182, Type 304 stainless steel Quickloc instrument nozzle to the Alloy 52 buttering on the alloy steel weld buildup nozzle. The applicant also provided information from the design drawing that the Quickloc instrument nozzle is welded to the Alloy 52/152 buttering after final post-weld heat treatment of the reactor vessel head. The staff finds that welding the Type 304 stainless steel Quickloc instrument nozzle after the final post-weld heat treatment of the reactor vessel closure head and Quickloc nozzle weld buildup minimizes the sensitization of the stainless steel Quickloc instrument nozzle. Therefore, the staff finds that the fabrication sequence of the Quickloc parts to the reactor vessel head using ASME Code, Section III ensures the integrity of the weld, including the final post-weld heat treatment of the reactor vessel head with the Quickloc nozzle weld buildup to ensure the material properties of the low alloy steel are retained.

In addition, the applicant stated that the Quickloc mechanism contains both pressure boundary and non-pressure boundary (internal structure) parts. The pressure boundary parts meet the requirements of ASME Code, Section III, Subsection NB, while the non-pressure boundary parts meet the requirements of ASME Code, Section III, Subsection NG. The staff finds the design and fabrication of the pressure boundary to be acceptable since it is evaluated in accordance with Class 1 requirements of ASME Code, Section III. The NRC staff also finds it appropriate for the design and fabrication of the non-pressure boundary parts to be designed and fabricated to the requirements of ASME Code, Section III, Subsection NG, which applies to reactor vessel internals and core support structures.

In regard to inspection during fabrication, the applicant provided the specific inspections to be performed for the pressure boundary parts in its response to RAI-SRP5.2.3-CIB1-03, Revision 0, dated January 15, 2010, and in response to RAI-SRP5.2.3-CIB1-03, Revision 1 dated April 21, 2010. These inspections included radiography and ultrasonic testing of the weld buttering and dissimilar metal weld. The weld buildup would also receive a magnetic particle inspection and an ultrasonic inspection, while the cladding and dissimilar metal weld would receive a liquid penetrant inspection. The inspections for the buttering and cladding would be performed in accordance with ASME Code, Section V, while the remainder of the inspections would be performed in accordance with Section III of the ASME Code, Subsection NB-5000. Since the fabrication inspections include volumetric examination of all the welds, and the inspections are in accordance with the ASME Code, the staff finds these inspections acceptable to ensure the integrity of the welds and base material.

However, to ensure the integrity of the welds is maintained, the NRC staff asked what type of inservice inspection is performed for these welds. In a letter dated January 15, 2010, the applicant stated that the dissimilar metal weld joining the stainless steel Quickloc instrument nozzle to the Alloy 52 buttering on the alloy steel Quickloc weld buildup requires an inservice inspection in accordance with Section XI of the ASME Code, Table IWB-2500-1, Item Number B5.10 (Category B-F). The NRC staff finds that Category B-F, Item Number B5.10 for dissimilar metal welds in pipe sizes larger than 10.2 cm (4 in) is the appropriate category for this weld to verify that the structural integrity is maintained as required by 10 CFR Part 50, Appendix A, GDC 30 and GDC 32, “Inspection of Reactor Coolant Pressure Boundary.”

In its response to RAI-SRP5.2.3-CIB1-03, Revision 1, the applicant stated that the Quickloc nozzle weld buildup is a base metal weld buildup, and is considered an extension of the reactor vessel head forging and, therefore, would require no inservice inspections. The NRC staff does not agree that inservice inspections are not required, since even a non-structural attachment weld to the reactor vessel, Examination Category B-K in the ASME Code, Section XI, requires a surface examination for inservice inspections. The staff also notes that the Quickloc nozzle weld buildup (maximum height of 37.03 cm (14.58 in) with a 19.1 cm (7.5 in) outside diameter and approximately 2.5 cm (1 in) thick) is designed as a nozzle penetration in the reactor vessel head, which forms the reactor coolant pressure boundary. Therefore, the overall dimensions of the weld buildup and the fact that this weld buildup serves as both a pressure boundary for the reactor coolant and a structural member to attach the Quickloc mechanism would necessitate that an inservice inspection be performed in order to verify that the structural integrity of the weld buildup is maintained as required by 10 CFR Part 50, Appendix A, GDC 30 and GDC 32.

In its response to RAI-SRP5.2.3-CIB1-03, Revision 3, dated August 3, 2010, the applicant stated that the inservice inspection of the Quickloc nozzle weld buildup is not a design issue and, therefore, should be addressed in the inservice inspection program to be developed by the COL applicant. Therefore, the applicant proposed a COL item in Section 5.3.6.6 of the AP1000 DCD, which states:

> The Combined License holder will establish an in-service inspection program prior to fuel load. The in-service inspection program will include the performance of a 100 percent volumetric examination of the weld build-up on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI. The weld buildup shall meet the acceptance standards of ASME Code, Section XI, IWB-3514. Personnel performing examinations and the ultrasonic examination systems shall be qualified in accordance with ASME Code, Section XI, Appendix VIII. Alternatively, the Combined License holder may develop an alternative inspection in conjunction with the voluntary consensus standards bodies (i.e., ASME) and submit to the NRC for approval.

The staff agrees that the COL applicant should include the appropriate inservice inspection for the Quickloc nozzle weld buildup, since Section XI of the ASME Code does not specifically address this type of weld buildup on the reactor vessel head. The NRC also agrees that a volumetric examination (ultrasonic) of all eight Quickloc nozzle weld buildups performed with procedures and personnel with similar qualifications (ASME Code, Section XI, Appendix VIII) for other Class 1 ASME Code welds in the reactor coolant pressure boundary provides assurance of the integrity of this reactor coolant pressure boundary weld. The staff also finds the acceptance criteria of IWB-3514 for Category B-F welds in Section XI of the ASME Code is the appropriate acceptance standard for detecting the type of flaws that could affect the integrity of the Quickloc nozzle weld buildup and the reactor coolant pressure boundary. The staff notes that the inspection of the Quickloc nozzle weld buildup can be performed in conjunction with the adjacent dissimilar metal weld using the same inspection procedures. As experience is gained on this weld, technical basis for alternative examinations can be developed through the voluntary consensus standards body process, such as ASME, and submitted for NRC approval. Therefore, the staff finds that this COL item adequately addresses the concern for inservice inspection of the Quickloc nozzle weld buildup, because it ensures that the COL applicant's inservice inspection program will include the appropriate inservice inspection for this weld, thereby assuring its integrity as required by 10 CFR Part 50, Appendix A, GDC 30 and GDC 32. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.2.3.2.4 Conclusions

In summary, the staff finds that the requested modification for the materials is acceptable because it satisfies the requirements in Section VIII B.5.b or Section VIII B.5.c of

10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design." The staff concludes that the changes to the pressure boundary materials in Revision 1 to TR-33 are technically acceptable. Section 4.5.1 of this report discusses the evaluation of the material changes for the CRDM components. The staff concludes that the TR-33, Revision 1, changes have been included in the AP1000 DCD are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.3 Reactor Vessel

5.3.2 Reactor Vessel Materials

5.3.2.1 Summary of Technical Information

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 5.3-3 (COL Action Item 5.3.2.2-1) by addressing the surveillance capsule lead factors and azimuthal locations in TR-23, "Surveillance Capsule Lead Factor and Azimuthal Location Confirmation," APP-GW-GLR-023, Revision 0, of September 2006. The applicant submitted TR-23 for staff review to demonstrate that it has met the requirements of COL Information Item 5.3-3. The submittal proposes a design change to locate the surveillance capsules in the minimum flux azimuthal locations to achieve lead factors in the range of 1.8 to 2.3, and as such satisfies the requirements in 10 CFR Part 50, Appendix H, "Reactor Vessel Surveillance Material Requirements," regarding surveillance capsule lead factors. The proposed change will eliminate the need for COL applicants to address the surveillance capsule lead factors and azimuthal locations requirement of COL Action Item 5.3.2.2-1.

In Section 5.3.6.3 of the AP1000 DCD, Revision 15, the COL action item states the following:

> The Combined License applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 5.3-3 by addressing the surveillance capsule lead factors and azimuthal locations in TR-23. The proposed revision to Section 5.3.6.3 of the DCD states the following:

> The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-023 (Reference 7), and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.
>
> The following words represent the original Combined License Information item commitment, which has been addressed as discussed above:
>
>> The Combined License applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

5.3.2.2 Evaluation

The surveillance capsule location in Figure 5.3-4 in Revision 15 of the DCD is at 29.70 degrees on either side of the 0- to 180-degree axis. The total number of surveillance capsules is eight. In this location, the estimated lead factor is outside (higher than) the range recommended in ASTM Standard E 185-82, “Standard Practice for Conducting Surveillance Test for Light-Water Cooled Nuclear Power Reactor Vessels,” incorporated by reference into Appendix H to 10 CFR Part 50. To resolve this issue, the applicant proposed to use the locations (at the azimuthal minimums of the neutron flux, E > 1.0 million electron volts) at 45, 135, 225, and 315 degrees. However, the 45-degree location cannot be used because of mechanical interference with reactor internals; therefore, it is proposed to have three capsules per location at the 135- and 315-degree locations and two capsules at 225 degrees.

The applicant performed an analysis for the anticipated range of lead factors at the proposed locations, and the results indicate that the values are in the range 1.8 to 2.3 (i.e., within the required values in Appendix H). The calculations were carried out for an equilibrium core power distribution and adhere to the guidance in RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” As stated in the guide, the cross-sections used (BUGLE-96) are based on the ENDF/B-VI file, the scattering cross-section approximation is the P_3 Legendre polynomial expansion, the angular quadrature approximation is S_8, and a 1/8 core symmetry was used. Because the 225-degree location contains only two capsules, two 1/8 core symmetry models were developed, one each for the two- and the three-capsule arrangements. Both models are based on the synthesis method; that is, the three-dimensional power distribution is derived from the synthesis of the (r, θ) and (r, z) distributions. The flux and fluence distributions derived in this manner are acceptable because the applicant followed the guidance in RG 1.190. This satisfies the requirements in GDC 14, “Reactor Coolant Pressure Boundary,” GDC 30, and GDC 31.

Section 5.3.6.3 of the DCD, Revision 17, references TR-23 in its entirety. Although the entire report can be referenced for additional information, the staff finds that for clarity and completeness of the DCD as a stand-alone document, the DCD should include portions of the TR describing important design details. The staff issued RAI-SRP5.3.1-CIB1-01 on February 19, 2008, asking the applicant to include the following information in the AP1000 DCD: (1) the azimuthal locations of the capsules (in degrees) and the basis for these locations; (2) the calculated lead factors; and (3) Figure 1, “Surveillance Capsule Azimuthal Location.” The applicant’s RAI response, dated March 28, 2008, appropriately described the changes to the DCD to incorporate information from the TR. These design details were incorporated into Section 5.3.2.6 and Figure 5.3-4 of Revision 17 and subsequent DCD revisions.

In addition, the submittal states that the applicant examined the presence of the surveillance capsules in the proposed locations and concluded that there is no interference with required potential actions to mitigate severe accidents.

The staff found that the DCD, Revision 16 changes discussed by the applicant in TR-23 meet the requirements of Appendix H to 10 CFR Part 50 and are acceptable. Therefore, AP1000 COL Information Item 5.3-3 is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the changes incorporated into Revision 17 contribute to the increased standardization of the certification information in the AP1000 DCD.

5.3.2.3 Conclusions

On the basis of its review of TR-23 and the associated changes to the DCD, the staff concludes that the proposed change meets the requirements of Appendix H to 10 CFR Part 50, and that the applicant has provided sufficient design information to close out COL Information Item 5.3-3. Furthermore, the staff finds that the TR-23 conclusions regarding the surveillance capsule lead factors and azimuthal locations are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.3.3 Pressure-Temperature Limits

5.3.3.1 Introduction

In Revision 16 of the AP1000 DCD, the applicant included generic bounding P/T limits. In a letter dated May 30, 2008, the applicant submitted APP-RXS-Z0R-001, "AP1000 Generic Pressure Temperature Limits Report," Revision 1, which describes the generic pressure-temperature limits report (PTLR) for the AP1000. The generic bounding P/T limits in Revision 17 to the DCD are the same P/T limits used in the generic PTLR. The applicant plans to have AP1000 COL applicants use the generic PTLR described in APP-RXS-Z0R-001, Revision 1, when developing their plant-specific P/T limits.

The applicant submitted TR-33 and supporting information in a letter dated June 7, 2007, to provide technical justification for the proposed changes. The staff's evaluation of the proposed changes is provided below.

5.3.3.2 Evaluation

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates P/T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988; and NUREG-0800 Section 5.3.2. RG 1.99 describes the methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron irradiation. Appendix G to 10 CFR Part 50 requires that P/T limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to ASME Code, Section XI.

a. Background

NUREG-0800 Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to ASME Code, Section XI. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. Appendix G to ASME Code, Section XI, requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on stress intensities resulting from hydrostatic testing. Appendix G to ASME Code, Section XI, also requires a safety factor of 1.0 on stress intensities resulting from thermal loads for normal and transient operating conditions as well as for hydrostatic testing. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress (i.e., of axial orientation). This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to six times its depth. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T limit curves are the 1/4 thickness (1/4T) and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to ASME Code, Section XI, requires that licensees determine the ART (or adjusted RT_{NDT}) at the 1/4T and 3/4T locations. The ART is defined as the sum, the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term.

RG 1.99 provides guidance on the determination of ΔRT_{NDT} and the margin term. ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculation procedures.

b. The Applicant's Evaluation

In RAI-TR33-001, dated April 23, 2007, the staff asked the applicant to provide the following information:

1. Detailed methodology used in the development of the P-T limit curves.
2. Beltline material properties (Cu, Ni) assumed, including initial RT_{NDT} of materials.
3. Fluence assumed in the calculation of adjusted of RT_{NDT}.
4. P/T data points.

The applicant provided this information in a letter dated June 7, 2007, since it plans for the curves to be used by the COL holders as bounding curves.

The methodologies provided by the applicant are consistent with those in Westinghouse Commercial Automatic Power (WCAP)-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 4 of May 2004. The P/T limits, which are valid for up to 54 effective full-power years (EFPYs) of operation, were calculated with chemistry factors obtained from the tables in RG 1.99. As discussed in Section 5.2.3.2.2 of this report, the applicant proposed (and the staff agrees with) changing the maximum wt% of copper to 0.06 percent. This is the value the applicant used for the calculations of AP1000 bounding P/T limit curves. The initial RT_{NDT} values are generic values for the design. Table 5-2 shows the values the applicant has provided for limiting ART calculations of the beltline girth welds at two locations.

Table 5-2. Values for Limiting ART Calculations of the Beltline Girth Welds

	1/4T Location	3/4T Location
Fluence at 54 EFPYs	1.510×10^{19}	5.51×10^{18}
Chemistry Factor	82	82
ΔRT_{NDT}	32.9 °C (91.3 °F)	20.2 °C (68.3 °F)

Table 5-2. Values for Limiting ART Calculations of the Beltline Girth Welds

	1/4T Location	3/4T Location
Initial RT_{NDT}	-28.9 °C (-20 °F)	-28.9 °C (-20 °F)
Margin	18.6 °C (65.5 °F)	18.6 °C (65.5 °F)
ART	58.3 °C (137 °F)	45.6 °C (114 °F)

In addition, the applicant provided P/T data points without margins for instrumentation errors. The staff used this information to evaluate the acceptability of the proposed bounding P/T limit curves.

c. Evaluation

As discussed previously, the applicant requested approval for bounding P/T limit curves to be used by COL holders with the AP1000 design. The staff performed independent calculations of the ART values using the methodology in RG 1.99. Based on these calculations, the staff verified the limiting ART values at the beltline girth welds.

Given the acceptability of the applicant's calculated ART values for the limiting beltline material to 54 EFPYs, the staff evaluated the P/T limit curves for acceptability by performing a finite set of check calculations based on information submitted by the applicant and by using the methodologies referenced in the ASME Code (as indicated in NUREG-0800 Section 5.3.2). The staff verified that the proposed P/T limit curves satisfy the requirements in Section IV.A.2 of Appendix G to 10 CFR Part 50.

In addition, Appendix G to 10 CFR Part 50 also imposes a minimum temperature at the closure flange region based on the reference temperature for the flange material. Section IV.A.2 of Appendix G to 10 CFR Part 50 states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature at the closure flange region, which is highly stressed by the bolt preload, must exceed the reference temperature of the material in that region by at least 71.1 °C (160 °F) for core critical operation, 48.9 °C (120 °F) for normal, noncritical core operation, and by 32.2 °C (90 °F) for hydrostatic pressure tests and leak tests. Based on this limiting flange reference temperature, the staff has determined that the proposed P/T limits have satisfied the above requirements for the closure flange region during all modes of normal operation and for hydrostatic pressure and leak testing.

Based on this independent assessment, the staff concludes that the applicant's proposed P/T limit curves meet the requirements of Appendix G to 10 CFR Part 50 and are acceptable for operation of an AP1000 design through 54 EFPYs of facility operation. This determination; however, is only valid for the material properties and the projected fluence identified in this SER. Any changes to these values will require additional review. Revision 17 to the AP1000 DCD includes these revised P/T limit curves.

These revised P/T limits are generic and may be used by COL holders referencing the AP1000 certified design, contingent upon verification by the COL holder of the material properties and fluence projection. At this time, the NRC has not issued a COL for any AP1000 plant. Therefore, the proposed P/T limits contribute to the increased standardization of the certification information in the AP1000 DCD.

In AP1000 DCD, Revision 15, COL Information Item 5.3.6.1, the applicant stated that COL applicants will address the use of plant-specific P/T limits. In TR-6, "AP1000 As-Built COL Information Items," APP-GW-GLR-021, Revision 0 of June 2006, the applicant proposed to change the responsibility from the COL applicant to the COL holder. However, in subsequent discussions between the NRC staff and the applicant, the applicant decided to provide a generic PTLR in conjunction with a future AP1000 DCD amendment for use by AP1000 COL applicants. Upon review and approval of the PTLR, COL applicants will be able to use it for their respective plants as long as the PTLR methodology remains the same. In conjunction with the Bellefonte reference COL application review, the staff is requesting the COL applicant provide license conditions in which the COL holder will be required to: (1) update its P/T limits using the PTLR methodologies approved in the AP1000 DCD and using plant-specific material properties; and (2) to inform the NRC of its plans to use updated P/T limits. The applicant provided its generic PTLR for NRC review and approval in a letter dated May 30, 2008. The NRC discussed its evaluation of the PTLR in a letter to the applicant dated December 30, 2008. However, the staff finds that the applicant needs to provide the PTLR reference in its DCD. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In reviewing the AP1000 PTLR, the staff noted one inconsistency between the AP1000 PTLR and Revision 17 to the DCD concerning the listed nickel content of the circumferential reactor vessel beltline weld. Table 3 of the AP1000 PTLR listed the maximum allowable nickel content as 0.95 percent by weight. However, Table 5.3-1 in Revision 17 of the DCD listed the maximum allowable nickel content as 0.85 percent. In RAI-SRP5.3.2-CIB1-01 dated July 31, 2008, the NRC asked the applicant to resolve this discrepancy by amending either the DCD or the AP1000 PTLR to specify the same maximum nickel content of 0.85 percent or 0.95 percent. In its response dated September 5, 2008, the applicant indicated that the actual limiting value of the nickel content of the reactor vessel beltline weld is 0.85 percent by weight. Furthermore, the applicant indicated that this limiting nickel content value would be changed in Table 3 of the AP1000 PTLR to be 0.85 percent, consistent with the value established in Section 5.3 of the AP1000 DCD, Revision 17. In a letter dated April 2, 2009, the applicant provided Revision 2 of the AP1000 PTLR, which includes the proposed changes. The staff finds that the applicant has appropriately addressed the RAI and has verified that Revision 2 of the AP1000 PTLR incorporates the proposed revision to the maximum nickel content. As a result, RAI-SRP5.3.2-CIB1-01 is closed.

Criterion 3 of PTLR (Reference generic letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits") requires the evaluation of the LTOP system. AP1000 Technical Specification Limited Condition for Operation (LCO) 3.4.14 does not specify the normal RNS suction relief valve lift pressure setpoint. The applicant should revise LCO 3.4.14 to state that the RNS suction relief valve lift setpoint be within the specified limit in the PTLR. In a letter dated May 20, 2009, the applicant provided a supplemental response to RAI-SRP5.3.2-CIB1-02, which includes a revision to the PTLR with the following addition:

> 2.1 Low Temperature Overpressure Protection (LTOP) System
>
> > The Normal Residual Heat Removal System (RNS) pump suction line relief valve will have a lift setpoint of 500 psig with a full open pressure of 550 psig. The lift setpoint has been developed using the NRC approved methodology specified in Reference 2.

From the review of the applicant's response, the staff concludes that the provisions in the response are acceptable because they adequately address PTLR Criterion 3 of GL 96-03. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.3.3.3 Conclusions

The staff finds that the proposed bounding P/T limit curves meet the requirements of Appendix G to 10 CFR Part 50 and are acceptable for the stated material properties and projected fluence for 54 EFPYs. Furthermore, the staff finds that the TR-33 conclusions regarding the bounding P/T limit curves are generic and are expected to apply to all COL applications referencing the AP1000 DC. The staff reviewed and approved the applicant's PTLR methodology for the AP1000. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.3.6 Reactor Vessel Insulation

The applicant has completed the design of the reactor vessel insulation system (RVIS). In TR-24, "Reactor Vessel Insulation System - Verification of In-Vessel Retention Design Bases," APP-GW-GLR-060, the applicant provided information to demonstrate that the RVIS is designed to provide adequate cooling to ensure in-vessel retention of a damaged and relocated core. On this basis, the applicant proposed to close COL Information Item 5.3-5 (COL Action Item 19.2.3.3.1.3.2-1). The staff's evaluation of the RVIS design is in Section 19.2.3.3.1.3.2 of this report.

5.4 Reactor Coolant System Component and Subsystem Design

5.4.1 Reactor Coolant Pump Assembly

5.4.1.1 Summary of Technical Information

In Section 5.4.1 of Revision 17 of the DCD, the applicant proposed to make changes related to the certified RCP design. These changes include: (1) change of the canned motor RCP design description to a more generic sealless pump; (2) use of an externally mounted heat exchanger; (3) change of the RCP flywheel design from depleted uranium to bimetallic construction; and (4) other miscellaneous changes.

5.4.1.2 Pump and Motor Design

The certified AP1000 RCP design, as described in Section 5.4.1.2.1, "Design Description," of Revision 15 to the DCD, is a single-stage, hermetically sealed, high-inertia, centrifugal canned motor pump. In TR-34, "AP1000 Licensing Design Change Document for Generic Reactor Coolant Pump," submitted by letter dated November 17, 2006, the applicant proposed to replace the specific term "canned motor pump" with a generic sealless pump design to provide flexibility in the selection of a specific pump design and pump vendors. In DCD Revision 17, the applicant revised Section 5.4.1.2.1 to replace the term "canned-motor pump" with the phrase "sealless pump of either canned motor or wet winding design." In addition, in many DCD Tier 2 sections (e.g., Sections 1.9.3, 1.9.4.2.3, 1.9.5.1.6, 3.5.1.2.1.4, 3.9.2.3, 4.4.4.6, 5.1.2, and 5.1.3.3 and Table 5.1-2) associated with the RCPs, the applicant replaced the term "canned-motor pump" with "sealless pump," or "sealless pump of canned motor design." In

DCD Tier 1, Section 2.1.2, “Reactor Coolant System,” the applicant replaced the term “canned motor reactor coolant pumps” in the design description with the phrase “sealless reactor coolant pumps.”

The canned motor pump design contains the motor and all rotating components inside a pressure vessel designed for full RCS pressure. Since the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. Thus, the concern regarding RCP seal failure is eliminated. The staff agrees that the use of the generic term “sealless pump” achieves the same objective as specifying a “canned motor pump” with regard to the seal failure concern. However, the NRC had based its review and acceptance of the AP1000 DCD on the canned motor pump design. In DCD Revision 16, the applicant did not provide materials related to the wet winding pump design.

Revised DCD Figure 5.4-1, “Reactor Coolant Pump,” and Reference 10 in DCD Section 5.4.16 are based on a canned motor pump. In the June 7, 2007, response to RAI-TR34-1, the applicant stated that the RCP design and licensing basis for the standard AP1000 plant is the canned motor RCP. In the October 5, 2007, response to RAI-TR34-SRSB-01, the applicant stated that for an applicant to use the wet winding sealless RCP, Reference 10 in Section 5.4.16 and Figure 5.4-1 of the DCD would need to be changed in the COL FSAR to incorporate the wet winding sealless design. This change would be a departure from the DCD and would open licensing review of the design change. Detailed design information on a wet winding sealless RCP design would be needed for NRC review and approval. In the April 11, 2008, response to RAI-SRP5.4.1-CIB-01, the applicant proposed changes to DCD Revision 16 to identify the canned motor design as the RCP design for the standard design, eliminating the mention of the wet winding motor design from the DCD. In DCD Revision 17, the applicant revised Section 5.4.1.2.1 to state that the RCP is a “single-stage, hermetically sealed, high-inertia, centrifugal sealless pump of canned motor design.” This sentence is re-classified as Tier 2* information in that staff approval is required before implementing a change in this information. Therefore, the AP1000 design basis for the RCP is a sealless pump of canned motor design. Use of a sealless pump design of another type, such as a wet winding motor design, would be a departure from the DCD and would require detailed design information in the COL FSAR for NRC review and approval. The staff finds this acceptable.

In DCD Tier 1, Section 2.1.2, “Reactor Coolant System,” the applicant also replaced the term “canned motor reactor coolant pumps” in the design description with the phrase “sealless reactor coolant pumps.”

5.4.1.3 Heat Exchanger Design

In DCD Revision 17, the applicant proposed to change the heat removal design of the RCP by using an externally mounted, conventional shell and tube heat exchanger and a stator cooling jacket to replace the existing thermal barrier internal cooling coils and wraparound heat exchanger configuration. The applicant revised DCD Sections 5.4.1.2.1 and 5.4.1.2.2 to describe the revised motor cooling arrangement. An auxiliary impeller at the lower part of the rotor shaft circulates a controlled volume of the reactor coolant through the motor cavity, where the rotor, bearing and stator are cooled, and through an external heat exchanger where the coolant is cooled to about 65 °C by the component cooling water (CCW) circulating on the shell side. The CCW also circulates through a cooling jacket on the outside on the motor housing to cool the stator. The applicant revised DCD Figure 5.4-1 to show the external heat exchanger configuration. In addition, it revised DCD Section 5.4.1.3.3, “Pressure Boundary Integrity,” to include the external piping and tube side of the external heat exchangers as a part of the pressure boundary components that meet the requirements of ASME Code, Section III.

In TR-34, the applicant explained that it changed to an external heat exchanger for the RCP because as the detailed design of the pump progressed, the increased heat transfer requirements on the heat exchanger resulting from increased motor power requirements and the effects of design transients on motor operation have resulted in significant manufacturing challenges associated with the wraparound heat exchanger design. Therefore, a conventional shell and tube heat exchanger mounted on the pump flange is implemented to replace the current wraparound heat exchanger.

In its June 7, 2007, response to RAI-TR34-2, the applicant summarized the RCP cooling design. This includes the heat source from motor electrical loss, fluid and friction losses, hot primary coolant crossing the thermal barrier into the motor, and the heat removal capacity of heat exchanger and water jacket. The external heat exchanger is specified to remove 2.4 megawatts with 540 gallons per minute (gpm) of CCW at 35 °C and 600 gpm of primary flow at 69 °C. In the October 5, 2007, supplemental response to RAI-TR34-SRSB-02, the applicant indicated that the RCP design specification defining the external heat exchanger design requirements is passed to the external heat exchanger supplier.

Establishment of the design requirements allows for the finalization of the design. The external heat exchanger generic design report will provide detailed design information for the external heat exchanger. In its July 3, 2008, response to RAI-SRP5.4.1-SRSB-01, the applicant indicated that this design report would be available for NRC review on October 31, 2008. In addition, each pump would be performance tested with the heat exchanger intended for field use before shipment. The applicant also indicated that the auxiliary impeller has been designed for an Euler head rise of 240 feet (ft) at 1782 revolutions per minute. Prototype RCP testing in the future will verify that the actual bearing water flow rate is sufficient to satisfy design requirements. If an unpredicted difference occurs between the calculated bearing water flow rate and the measured test value, an easily implemented design change of adding an annular ring to the motor shaft may be executed to increase auxiliary impeller flow capacity.

The staff identified review and acceptance of the external heat exchanger design report as Open Item OI-SRP5.4.1-SRSB-01.

In a letter dated March 26, 2010, the applicant submitted a response to Open Item OI-SRP5.4.1-SRSB-01 that included the external heat exchanger design specifications regarding its functions and design requirements. The design specifications specify the objective of the external heat exchanger to cool primary side water used in a RCP motor, and specify that the primary-side and secondary-side components of the heat exchanger shall meet Section III, Division 1, Class 1 and Class 3, respectively, of the ASME Code. In addition, the external heat exchanger shell will serve as supports for Section III, Class 1 heat exchanger tubes that form part of the primary pressure boundary, and as such cannot be allowed to fail in such a way as to prevent the Class 1 pressure boundary from performing its function. The heat exchanger design requirements specify the heat exchanger configuration and nozzle size, the design basis life of 60 years, normal operating conditions, allowable pressure drop of the primary and secondary sides, design pressure and temperature, and thermal transient operating conditions, material, and vibration analysis. Regarding the vibration analysis, it specifies that the heat exchanger shall be designed to avoid flow-induced tube vibration under maximum expected flows and design temperature conditions. The adequacy with respect to flow-induced vibration shall be demonstrated by either: (a) an identical heat exchanger having operated satisfactorily under flow and temperature conditions at least as severe as those expected for the heat exchanger; or (b) the maximum expected secondary side flow velocities will be below the critical velocities for fluid-elastic excitation, and the lowest tube natural frequency exceeds by a factor of 1.5 the vortex shedding frequency calculated for the maximum expected secondary side flow velocity. Based on this review, the staff concludes that the applicant has successfully addressed the open item; therefore, Open Item OI-SRP5.4.1-SRSB-01 is closed.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

5.4.1.4.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant proposed to change the RCP design from a canned motor RCP with a depleted uranium flywheel to a generic sealless RCP with a bimetallic flywheel assembly. TR-34 identified changes to the design of the RCP and the technical justification for the proposed changes.

The AP1000 DCD, Revision 15, had specified a canned motor RCP with a depleted uranium flywheel. The Revision 16 changes to the AP1000 DCD specify a generic sealless RCP with a bimetallic flywheel assembly. Changing to a generic sealless RCP will provide flexibility in selecting a specific pump design and thus increase the number of possible pump vendors. Revision 16 to the AP1000 DCD includes a revised flywheel assembly design of bimetallic construction.

Concerning the material specifications for the flywheel, TR-34 stated that the preliminary flywheel design (in Revision 15 of the AP1000 DCD) employed an upper and lower flywheel assembly constructed of forged depleted uranium disks fitted to an inner stainless steel hub, which was fitted to the motor shaft. Structural integrity of the flywheel assemblies relied upon the strength of the depleted uranium forged disks. The depleted uranium flywheel was designed to meet the minimum rotating inertia value, 695.3 kilograms - square meter (kg-m^2) (16,500 pounds - square foot (lb-ft^2)), given in DCD Tier 2, Table 5.4-1. As the design progressed, it was determined that this inertia value needed to be increased to meet the pump coastdown used in the safety analyses as given in DCD Figure 15.3.2-1. To achieve the required inertia, the depleted uranium flywheel design required increases in length and/or diameter. However, increases in diameter resulted in stress levels beyond the design criteria limits and increases in length resulted in violation of the RCP space envelope as well as unacceptable rotor dynamics. Therefore, a revised flywheel assembly design of bi-metallic design with heavy tungsten alloy inserts was developed.

In a letter dated September 28, 2007, the applicant provided additional information concerning the materials and design of the flywheel assembly. The design features heavy tungsten alloy annular or cylindrical segments, which are machined and fitted around a central Type 403 stainless steel hub. The segments are held in place by an interference fit of an 18-Ni maraging steel retainer cylinder placed over the outside of the assembly. Alloy 690 endplates and an outer thin shell hermetically seal the assembly from primary coolant. Structural integrity of the flywheel assembly relies upon the stainless steel hub and the retainer cylinder. Both the upper and lower flywheels are of the same design.

In a letter dated June 12, 2007, the applicant submitted TR-106, “AP1000 Licensing Design Changes for Mechanical Systems and Component Design Updates,” which revised the flywheel enclosure (endplates and outer thin shell) materials. Design change 245 in TR-106 stated that the applicant would revise Section 5.4.1.3.6.3 of the AP1000 DCD by replacing references to

“Alloy 690” with the phrase “corrosion resistant material” for the flywheel enclosure (endplates and outer thin shell) to meet the intent of creating a generic RCP design description that would facilitate other future RCP suppliers. In addition, the applicant stated that the RCP supplier had proposed the material change from Alloy 690 to Alloy 625 because of the low coefficient of thermal expansion and better weldability of Alloy 625. In a letter dated October 5, 2007, the applicant provided supporting information concerning the test results and experience of using Alloy 625 in contact with the reactor coolant. The applicant also concluded that the AP1000 DCD should not include such detailed design information as referencing the specific material (i.e., Alloy 625); therefore, the AP1000 DCD would only specify “corrosion-resistant material.”

5.4.1.4.2 Evaluation

GDC 4 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. GDC 1 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards to ensure a quality product in keeping with the required safety function. The staff’s view of the Revision 16 changes related to Section 5.4.1.3.6.3 of the DCD is that they minimalize the potential for RCP flywheel failures and that the materials are adequate to ensure a quality product commensurate with the importance to safety.

Section 5.4.1.2.1 of the AP1000 DCD, Revision 16, changes the pump design from standard (canned motor) to a more generic “sealless” pump design (canned motor, wet-winding, etc.). The applicant revised Section 5.4.1.3.6.3 of the AP1000 DCD, Revision 16, to state that the analysis to determine the capacity of the housing to contain the fragments of the bi-metallic flywheel appears in Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, “Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel,” dated October 2006. This Curtiss-Wright report is only applicable for the canned-motor design. In RAI-SRP5.4.1-CIB1-01, the staff asked the applicant to change the DCD to provide the following:

- A specific RCP/flywheel design (i.e., a single-stage, high-inertia, centrifugal, sealless RCP of canned-motor design). Currently, this is the only RCP design that has a supporting analysis for the flywheel integrity and missile generation.

- The material specifications for the flywheel and the specific inspections to be performed on the flywheel.

- The material type for the end plates and outer shell since TR-106 replaced the material type “Alloy 690” with “corrosion resistant” from AP1000 DCD, Revision 16, Section 5.4.1.3.6.3.

In a letter dated April 11, 2008, the applicant responded to RAI-SRP5.4.1-CIB1-01 and provided the following:

- RCP design and flywheel structural analysis: AP1000 DCD, Revision 16 will be changed to incorporate the standard design RCP (a single-stage, hermetically sealed, high-inertia, centrifugal, sealless pump of canned-motor design).

- Flywheel material and inspections: information identifying the materials used in the flywheel, and the inspections that will be performed on the flywheel.

- Flywheel enclosure material: information identifying the materials used in the flywheel enclosure.

The staff reviewed this information and provides the following summary and evaluation of each issue.

Flywheel Material and Inspections

The Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, dated October 2006, used the material properties from the material specifications for the flywheel (Type 403 stainless steel inner hub—ASTM A336, Grade F6, 18-Ni maraging steel outer hub—AMS 6519, Vascomax T250 (UNS 6075) and heavy tungsten inserts—ASTM B777, Class 4) to evaluate the flywheel integrity and ability to minimize missile generations. AP1000 DCD Section 5.4.1.3.6.3 also relies on the material specifications and preservice nondestructive testing to demonstrate the integrity of the flywheel and justify the removal of the flywheel from an inservice inspection program. Therefore, to ensure structural integrity of the flywheel, as evaluated in the Curtiss-Wright report, the applicant should change the AP1000 DCD to include these material specifications or reference the Curtiss-Wright report for the material specifications to be used. The staff previously addressed this in RAI-SRP5.4.1-CIB1-01. The staff identified this as Open Item OI-SRP5.4.1-CIB1-01.

In a letter dated May 26, 2009, in response to RAI-SRP5.4.1-CIB1-01, Revision 1, the applicant stated that the flywheel outer hub material was changed from 18-Ni maraging steel to 18Mn-18Cr alloy steel (ASTM A289, Grade 8) as a result of lessons learned during a flywheel mockup assembly, and that the 18Mn-18Cr alloy steel is not susceptible to stress corrosion cracking and hydrogen embrittlement. A proposed change to Section 5.4.1.3.6.3 of the AP1000 DCD to include the new material specification was also provided in this letter. The staff confirmed that this material is more resistant to stress corrosion cracking than 18-Ni maraging steel based on the current operating experience of 18Mn-18Cr alloy steel retaining rings on generators used since the mid 1980's. The generator environment is more aggressive due to the hydrogen cooling and the wet oxygenated environment compared to the PWR reactor coolant water, which controls the oxygen content. Also, the staff notes that this alloy steel has a high chromium content, which is similar to stainless steels currently used in the reactor coolant. The 18Mn-18Cr material is similar to a high strength stainless steel with yield strength of 195 kips per square inch (ksi) compared to 250 ksi for the 18-Ni maraging steel. Since this alloy steel is not a nickel based alloy, such as Alloy 600, primary water stress corrosion cracking is not a concern. The NRC also notes that the 18Mn-18Cr alloy steel outer hub will be enclosed in an Alloy 625 flywheel enclosure to prevent the outer hub from contacting the reactor coolant. Therefore, the staff finds the use of the 18Mn-18Cr alloy steel acceptable based on the current operating experience of this material in an aggressive stress corrosion environment, and even if the flywheel ruptures, the flywheel would be contained in the RCP as discussed below, thereby preventing a missile that could impact safety-related equipment or structures, which meets the requirements of GDC 4.

In a letter dated August 4, 2009, in response to RAI-SRP5.4.1-CIB1-01, Revision 2, the applicant provided the material specifications for the inner hub (ASTM A336, Grade F6) and the tungsten inserts (ASTM B777, Class 4) in a proposed change to Section 5.4.1.3.6.3 of the AP1000 DCD as requested by the staff, which partially resolves Open

Item OI-SRP5.4.1-CIB1-01. In a subsequent revision to the AP1000 DCD the material specifications as stated in letters dated May 26, 2009 and August 4, 2009 were included, which resolves this issue.

However, since the outer hub material was changed to 18MNn-18Cr, the evaluation of the flywheel in Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, dated October 2006, is no longer valid since it does not bound the new material (18Mn-18Cr). In a letter dated August 4, 2009, the applicant submitted a revised flywheel analysis, Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, Revision 2, dated July 2009, which included the new material (18Mn-18Cr), and is, therefore, part of Open Item OI-SRP5.4.1-CIB1-01, which is discussed in the "RCP design and flywheel structural analysis" section below.

In regard to the preservice inspection of the flywheel, the applicant provided changes to AP1000 DCD, Revision 17, Section 5.4.1.3.6.3, to include surface examinations and volumetric inspections of the inner hub and retainer cylinder ring, and an overspeed spin test followed by a visual inspection and leak test of the final assembly in accordance with ASME Code, Section III. In addition, impact/fracture toughness testing will also be conducted on the inner hub and retainer cylinder ring. The staff finds that these inspections provide reasonable assurance of the integrity of the flywheel during fabrication and ensure that the basis for safe operation of the RCP will be maintained.

Flywheel Enclosure Material

Design change 245 in TR-106 stated that the applicant would revise Section 5.4.1.3.6.3 of the AP1000 DCD by replacing references to "Alloy 690" with the phrase "corrosion resistant material" for the flywheel enclosure (endplates and outer thin shell) to meet the intent of creating a generic RCP design description. The NRC staff notes that Section 5.4.1.3.6.3 of the AP1000 DCD credits the use of the flywheel enclosure to prevent contact with the reactor coolant and to minimize the potential for corrosion of the flywheel and contamination of the reactor coolant. A leak in the flywheel enclosure during operation could result in an out-of-balance flywheel assembly. In addition, the applicant stated that the use of the required material specifications and nondestructive testing during fabrication of each flywheel demonstrates the quality of the flywheel, thereby permitting the removal of the requirement for periodic inservice inspections of the flywheel to ensure that the basis for safe operation of the RCP is maintained. Based on the above, the NRC staff notes that, although the flywheel enclosure is not credited for retaining potential flywheel missiles, the flywheel enclosure is an integral part of ensuring the integrity of the flywheel so that it will not generate a missile. In the course of its review, the staff identified that Section 5.4.1.3.6.3 of the AP1000 DCD should be revised to state the material type (i.e., Alloy 690 and/or Alloy 625) for the flywheel enclosure.

In a letter dated April 11, 2008, the applicant included Alloy 625 in Section 5.4.1.3.6.3 of the AP1000 DCD as the material used for the flywheel enclosure. Therefore, the NRC staff considers Alloy 625, as referenced in ASTM B-443, "Standard Specification for Nickel-Chromium-Molybdenum-Columbium Alloy (UNS N06625) and Nickel-Chromium-Molybdenum-Silicon Alloy (UNS N06219) Plate, Sheet, and Strip," and ASTM B-564, "Standard Specification for Nickel Alloy Forging," to be an acceptable material for the flywheel enclosure, based on current operating experience in fuel assemblies and testing performed by Bettis Atomic Power Laboratory, which was discussed in the applicant's letter dated October 5, 2007. With the inclusion of the material specification for the flywheel enclosure, the staff finds the change to Section 5.4.1.3.6.3 acceptable.

RCP Design and Flywheel Structural Analysis

Concerning the RCP design, see Section 5.4.1.2 of this report for the staff's evaluation. The staff concludes this standard design type (a single-stage, hermetically sealed, high-inertia, centrifugal, sealless pump of canned-motor design) is supported by an applicable flywheel integrity analysis, and it standardizes the certification information for all COL applicants.

The structural analysis of the bimetallic flywheel with heavy tungsten alloy inserts for the sealless RCP with canned motor and including a missile containment evaluation of a fractured flywheel was initially documented in Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, issued October 2006.

However, due to a change in material (18Mn-18Cr) for the outer hub of the flywheel, the applicant submitted a revised flywheel analysis, Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, Revision 2, dated July 2009, in a letter dated August 4, 2009. The August 4, 2009, letter also provided a proposed change to Section 5.4.16 of the AP1000 DCD, which references this new Curtiss-Wright analysis. This analysis used the material properties of 18Mn-18Cr, using a similar methodology as the previous analysis. However, the NRC staff noted that there were other changes to the RCP including material changes to the pump, dimensional changes to the flywheel, and an additional pump part (stator closure ring) adjacent to the upper flywheel that would contain the upper flywheel if it were to rupture. In addition, a missile generated from a fractured portion of the upper flywheel would now have to penetrate three parts (stator closure, stator closure ring, and thermal barrier) which are bolted together. Therefore, the staff identified this as part of Open Item OI-SRP5.4.1-CIB1-01, Revision 1, in that the analysis should also account for shearing of the bolts connecting these three parts.

In response to Open Item OI-SRP5.4.1-CIB1-01, Revision 1, in a letter dated June 6, 2011, the applicant stated that the new analysis assumes that the flywheel is contained by the three parts (stator closure, stator closure ring, and thermal barrier) which are bolted together, and act as one unit. These three parts act as one unit due to the combination of large compressive stresses from the main flange bolts and the recessed fits between the three parts, which transfer the shear load from one part to another. The NRC staff notes that the analysis is conservative since only a portion of the entire volume of the three parts (stator closure, stator closure ring, and thermal barrier) was credited in the analysis with respect to containing a fractured flywheel. Therefore, the NRC staff finds that the shear mechanism was analyzed and bounded by the analysis. In addition, Revision 2 of the analysis accounted for the dimensional changes in the flywheel, and the outer hub material properties of 18Mn-18Cr.

In response to RAI-SRP5.2.3-01, Revisions 1 and 2, dated December 31, 2009, and April 7, 2010, respectively, the applicant provided information on material changes to carbon steel for specific parts of the pump, which are the stator main flange, stator shell, and external heat exchanger supports. These parts are not exposed to reactor coolant and are not credited for containing a fractured flywheel. Therefore, the NRC staff agrees that these material changes to the stator main flange, stator shell, and external heat exchanger supports have no effect on the flywheel analysis. In addition, the staff finds the use of carbon steel for these parts acceptable, since they are not exposed to the reactor coolant. Therefore, the staff finds Open Item OI-SRP5.4.1-CIB1-01 completely resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The flywheel analysis, Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, Revision 2, issued July 2009, demonstrates that the calculated stresses during both normal operating conditions and design conditions are less than the applicable stress limits. In addition, missile penetration calculations show that in the unlikely event of a flywheel fracture, the flywheel assembly components will not have sufficient energy to penetrate the pump pressure boundary structures. Therefore, the NRC staff finds the structural analysis for the bi-metallic flywheel with heavy tungsten alloy inserts acceptable because the missile penetration calculations show that if the flywheel fractures, the flywheel assembly components will not have sufficient energy to penetrate the pump pressure boundary structures; therefore, the design meets the requirements of GDC 4. The NRC staff had approved the previous AP1000 canned motor design with a depleted uranium flywheel assembly, which also relies on the pump casing to confine the flywheel if the flywheel fractures. The missile penetration calculations for both material designs used similar methodologies.

5.4.1.4.3 Conclusion

Based on the above evaluation, the staff finds that the material specifications used and the preservice inspections provide reasonable assurance of the flywheel integrity, and that the Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009 demonstrates that if the flywheel assembly fails, the flywheel components will not penetrate the pump pressure boundary structures. Therefore, the staff finds the RCP flywheels acceptable since they meet the requirements of GDC 1, GDC 4, and ASME Code, Section III.

The NRC staff reviewed the changes in Section 5.4.1.3.6.3 of the AP1000 DCD and finds that the AP1000 DCD adequately incorporates the proposed changes, as identified in TR-34 and the RAI responses. Furthermore, the staff finds that these changes are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.4.1.5 Other Changes

In Revision 17 to the DCD, the applicant also proposed to change the following RCP design parameters in Table 5.4-1: (1) maximum continuous CCW inlet temperature; (2) motor/pump rotor minimum required moment of inertia; and (3) RCP estimated unit overall height and weight.

The applicant revised DCD Table 5.4-1 by adding Note 1, associated with the maximum continuous CCW inlet temperature of 95 °F, stating that an elevated CCW supply temperature of up to 110 °F may occur for up to 6 hours. In its response to RAI-SRP5.4.1-SRSB-02, the applicant explained that it intended Note 1 to clarify that although the design temperature of the CCW system cooling water to the RCP is 95 °F, transient conditions during normal plant cooldown or extreme climate events may occur during which the CCW system temperature may exceed 95 °F (but no more than 110 °F) for a short time. The staff concludes that the addition of Note 1 is acceptable.

In DCD Table 5.4-1, the applicant changed the motor/pump rotor minimum required moment of inertia from 16,500 lb-ft^2 to the phrase “Sufficient to provide flow coastdown as given in Figure 15.3.2-1.” The applicant had previously proposed a similar change to the acceptance criterion of the RCP rotating inertia in DCD Tier 1, Table 2.1.2-4, “Inspections, Tests, Analyses, and Acceptance Criteria,” to a specific pump coastdown curve specified in DCD Tier 1,

Figure 2.1.2-2, “Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down,” which is consistent with DCD Tier 2, Figure 15.3.2-1. The applicant proposed this change because the specified pump rotating inertia does not guarantee that the flows produced during pump coastdown will satisfy the analyses in the absence of information on pump resistance. The staff had found this change to be acceptable, as described in Supplement 1 to NUREG-1793. Therefore, the proposed change to Table 5.4-1 regarding pump rotor minimum required moment of inertia is made to be consistent with the four-pump coastdown curve in DCD Figure 15.3.2-1, which is used in the safety analysis of the design-basis transient of a complete loss of reactor coolant flow described in DCD Section 15.3.2; therefore, it is acceptable.

In Table 5.4-1, the applicant changed the RCP estimated unit overall height from 6.693 m (21 ft 11.5 in) to 6.706 m (22 ft), and it changed the total estimated weight of the motor and casing from 83688 kg (184,500 pounds mass (lbm)) to 90718 kg (200,000 lbm). These changes reflect the changes in the pump casing discharge nozzle and the use of the bi-metallic flywheel design.

In DCD Section 5.4.1.2.1, the applicant changed the RCP vibration monitoring system instrumentation from three-axis monitoring to five vibration monitors. In TR-34, the applicant stated that the instrumentation of the RCP is modified from the preliminary design to provide a more robust monitoring and diagnostic capability of the RCP. These changes include the addition of key phasors to aid in diagnostics in the event of high-vibration indications, and the addition of more vibration monitors to supply measurements in two planes at two different axial locations for diagnostic purposes. The staff concludes that these RCP instrumentation changes provide a more robust monitoring and diagnostic capability of the RCP without affecting RCP performance; therefore, they are acceptable.

In Revision 17 of DCD, the applicant updated Section 5.4.1.3.6.1 to specify the minimum damped natural frequency of the RCP rotating assembly as greater than 120 percent of the normal operating speed. The staff reviewed the various factors that were considered in determining the damped natural frequency of the RCPs. There is considerable energy dissipation from the effects of bearing films, can or winding annular fluid interaction, motor magnetic phenomena, and pump structure. The determination of the damped natural frequency of the RCP rotor bearing system model considered these effects on the damped natural frequency. The high degree of damping as a result of energy dissipation ensures a stable and smooth operation of the pump over a sufficiently wide range of pump operating speeds. The pumps have been analyzed for the response of the rotor and stator to external forcing functions. The analysis considered the support and connection of the pump to the SG and piping. The applicant evaluated the responses using criteria that included critical loads, stress deformation, and wear and displacement limits to establish the actual system critical speeds. The staff finds the evaluation criteria for determining the RCP damped natural frequency and critical speeds to be reasonable and acceptable. The staff also finds that there is high energy dissipation because of the various damping factors, which would ensure stable and smooth pump operation.

5.4.1.6 Conclusions

The staff has reviewed the proposed changes to the RCP design. Based on the evaluation described, the staff concludes that the AP1000 RCP design meets the requirement in GDC 10, “Reactor Design”; therefore, it is acceptable. The staff finds that the material specifications used and the preservice inspections provide reasonable assurance of the flywheel integrity, and that the Curtiss-Wright report demonstrates that, if the flywheel assembly fails, the flywheel components will not penetrate the pump pressure boundary structures. Therefore, the revisions proposed by the applicant to AP1000 DCD, Section 5.4.1, meet the requirements of GDC 1 and 4 and ASME Code, Section III and are acceptable.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Design

5.4.2.1.1 Summary of Technical Information

In Revision 17 to the AP1000 DCD, the applicant proposed changes to its SG design. In a letter dated November 29, 2006, the applicant submitted for staff review TR-35, Revision 0, "AP1000 Steam Generator Description Changes" (APP-GW-GLN-010), which provides the technical justification for the proposed SG design changes. Additional information regarding TR-35 appears in letters dated June 7, 2007, and September 7, 2007. The proposed design changes are described below.

Antivibration Bar

The original design of the antivibration bar is described in DCD Section 5.4.2.4.2 and detailed in DCD Figure 5.4-2. In Revision 17, the applicant revised the description by removing the words "wide strips of" from Section 5.4.2.4.2 and reconfiguring Figure 5.4-2 to show a general antivibration bar design.

Tube Expansion

DCD Sections 5.4.2.2, 5.4.2.4.1, and 5.4.2.4.2 describe the tube expansion process as a hydraulic expansion through the full depth of the tubesheet. The intent remains to achieve full depth expansion; however, the requirement to hydraulically expand through the full depth limits the manufacturing processes available for this expansion. In Revision 16, the applicant removed the reference to hydraulic expansion from DCD Sections 5.4.2.2, 5.4.2.4.1, and 5.4.2.4.2.

Primary Separator Design

The original design of the moisture separators is described as a 19.1cm (7.5-in) separator arrangement. The applicant determined that this design could not achieve the industry standard upper bound limit for moisture carryover (0.1 percent) and, therefore, changed the design to a 50.8-cm (20-in) separator arrangement. In Revision 17, the applicant revised Figure 5.4-2 to illustrate its typical 50.8-cm (20-in) separator arrangement.

Primary Separator Material

DCD Section 5.4.2.4.1 states, "Nickel-chromium-iron alloy in various forms is used for parts where high velocities could otherwise lead to erosion/corrosion. These include the nozzles on the feedwater ring, startup feedwater sparger, and some primary separator parts." In Revision 16 to the DCD, the applicant removed the phrase "and some primary separator parts."

Startup Feedwater Elevation

DCD Section 5.4.2.2 describes the startup feedwater nozzle elevation as being “just below” the main feedwater nozzle. In Revision 17 to the DCD, the applicant revised this to state that the startup feedwater nozzle elevation is the same as the main feedwater nozzle elevation.

Table 5.2-1 Correction

DCD Section 5.4.2.4.1 includes the table for the list of materials and is called “Table 5.2.3-1.” In Revision 16 to the DCD, the applicant revised the table number to “Table 5.2-1.”

Tube Location Identification

DCD Section 5.4.2.5 describes the number of tubes being scribed as “large.” However, the space between the tubes and welds is not sufficient to allow for scribing a large fraction of the tubes. In Revision 17 to the DCD, the applicant removed the word “large” from Section 5.4.2.5.

5.4.2.1.2 Evaluation

GDC 32 requires, in part, that the designs of all components that are part of the reactor coolant pressure boundary permit periodic inspection and testing of critical areas and features to assess their structural and leak-tight integrity. The staff reviewed changes to this section as related to the proposed SG design changes.

In TR-35, the applicant proposed design changes to its SGs. Specifically, the applicant made minor description changes to the following: antivibration bars, tube expansion process, primary separator design, primary separator material, startup feedwater elevation, and tube location identification. The staff finds that these proposed changes meet the requirements of GDC 32 and are acceptable because they are consistent with current industry designs and practices and do not present a challenge to the integrity of the SG. Some of the changes were necessary to correct erroneous statements and inaccurate descriptions.

The NRC staff reviewed the proposed changes to the AP1000 DCD. The proposed changes establish the proposed design as the single, standard design for all AP1000 plants. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD.

5.4.2.1.3 Conclusions

Based on the above evaluation, the staff concludes that the SG design changes meet GDC 32 and are acceptable because they are consistent with current industry designs and practices and do not present a challenge to the integrity of the SG. Furthermore, the staff finds that the conclusions regarding the proposed SG design changes are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.4.2.2 Steam Generator Inservice Inspection

5.4.2.2.1 Summary of Technical Information

In Revision 17 to the AP1000 DCD, proposed changes to the AP1000 generic Technical Specification (TS) related to adopting Technical Specification Task Force Traveler (TSTF)-449, “Steam Generator Tube Integrity,” Revision 4 of May 6, 2005. The current Westinghouse Owners Group Standard Technical Specifications (STS), NUREG-1431, “Standard Technical Specifications Westinghouse Plants,” Revision 3.1, dated December 1, 2005, incorporates TSTF-449. These changes also relate to COL Information Item 5.4-1 identified in DCD Section 5.4.15. Implementation of TSTF-449 includes the following TS changes:

- clarification to the definition of identified leakage
- changes to TS 3.4.7, “Reactor Coolant System Operational Leakage,” which modifies two condition statements and the two associated surveillance requirements
- addition of new TS 3.4.18, “Steam Generator (SG) Tube Integrity”
- modification of TS 5.5.4, which is renamed, “Steam Generator (SG) Program”
- modification of TS 5.6.8, “Steam Generator Tube Inspection Report”
- associated changes to the TS bases

5.4.2.2.2 Evaluation

GDC 32 requires, in part, that components that are part of the reactor coolant pressure boundary shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The staff evaluated the applicant’s proposed changes to the AP1000 generic TS as they relate to adopting TSTF-449 to ensure consistency with the latest revision of the STS using the acceptance criteria in NUREG-0800 Section 5.4.2.2, “Steam Generator (SG) Program.” In addition, 10 CFR 50.55a(b)(2)(iii) states that if the TS include SG surveillance requirements that are different than those in Article IWB-2000 of ASME Code, Section XI, then the SG tube inspection requirements are governed by the TS.

Accordingly, the staff compared the proposed changes to the corresponding language in the STS (NUREG-1431, Revision 3.1). The staff determined that the changes are consistent with the approved STS and are appropriate to leakage and SG tube integrity requirements as they apply to the AP1000 standard plant design. Therefore, they are acceptable. With respect to tube integrity considerations, the Model Delta-125 SG planned for the AP1000 closely resembles the Model Delta-75 SGs installed as replacements at some operating plants. The staff also confirmed that Revision 17 to the DCD retains Section 5.4.15, which requires COL applicants to address SG tube integrity with a SG tube surveillance program and periodic monitoring of degradation of SG internals. .

5.4.2.2.3 Conclusions

Based on the evaluation above, the staff concludes that the changes proposed to the AP1000 TS for SG tube integrity are acceptable and meet the requirements of GDC 32. This conclusion is based on the consistency of the SG tube integrity program with the STS for the applicant's domestic PWRs. Furthermore, the staff finds that the conclusions regarding the changes to the TS for SG tube integrity will apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.4.4 Main Steam Line Flow Restriction

5.4.4.1 Summary of Technical Information

In Revision 17 of DCD Section 5.4.4, two changes are made regarding the main steam line flow restrictor.

- In Section 5.4.4.2, the material for the steam generator flow restrictor venturi inserts is changed from nickel-chromium-iron Alloy 600 (ASME 163) to nickel-chromium-iron Alloy 690.
- In Section 5.4.4.3, the pressure drop of the flow restrictor at 100 percent steam flow is changed from approximately 8.0 pounds per square inch (psi) to 15 psi.

5.4.4.2 Evaluation

For Section 5.4.4.2, since Alloy 690 is used in the AP1000 SG design, the change of the flow restrictor venturi insert material from Alloy 600 to Alloy 690 is consistent with the SG design and is acceptable.

For Section 5.4.4.3, the pressure drop change is based on the design analysis and is acceptable.

5.4.4.3 Conclusion

The staff has reviewed the proposed changes to DCD Section 5.4.4. Based on the above evaluation, the staff concludes that the proposed changes are acceptable.

5.4.5 Pressurizer

Section 5.4.5 of DCD Revision 17 refers to TR-36, "AP1000 Pressurizer Design" APP-GW-GLR-016 of May 2006. In TR-36, the applicant summarizes the design changes for the AP1000 pressurizer to accommodate the available space as a result of piping analysis. The pressurizer design changes preserve the same total internal volume of 59.5 cubic meter (m^3) (2100 cubic feet (ft^3)), while reducing the vessel height from 1541.8 cm (607 in) to 1277.6 cm (503 in) and increasing the vessel diameter from 228.6 cm (90 in) to 254.0 cm (100 in). Though DCD Revision 17, Table 5.1-2 reflects the dimensional change, Section 5.4.5 does not, because the total pressurizer volume did not change. However, DCD Revision 17, Section 15.0 did change, because the changes to the pressurizer vessel cross-section area and vessel height affect the pressurizer water level and setpoints of several functions of the reactor protection system and engineered safety feature actuation systems. Chapter 15 of this SER addresses the effects on the transient and accident analyses.

5.4.7 Normal Residual Heat Removal System

5.4.7.1 Summary of Technical Information

The AP1000 RNS is a nonsafety-related system and is not required to operate to mitigate design-basis events. The primary functions of the RNS include the following: (1) remove decay heat and sensible heat from the core and the RCS; (2) provide LTOP; (3) provide a flow path for long-term, post accident makeup to the containment inventory; and (4) provide RCS and refueling cavity flow to the chemical and volume control system (CVS) for purification.

In Revision 17 to the DCD, the applicant proposed the following changes related to the RNS design:

- Revise the method of flow control through the RNS heat exchangers (HX) to improve the cooldown rate control.
- Provide auto close to V029, RNS motor-operated flow control valve to CVS, on high temperature to prevent potential damage to CVS demineralizer resin.
- Resize the RNS suction line relief valve to reduce valve instability.
- Increase ambient wet bulb temperature to 86.1 °F with zero percent exceedance to facilitate consideration of other potential nuclear plant sites.
- Relocate the RCS hot leg RTDs to provide required post accident monitoring data to upstream of the HX to indicate the RCS hot-leg temperature when in reduced inventory conditions.
- Revise the RNS long term makeup flow path to provide a simplified flow path for containment makeup through the manual containment isolation test connection valve in the discharge of the RNS.
- Revise Figure 5.4-7 to reflect containment penetration valve and piping changes.

5.4.7.2 Evaluation

The staff reviewed the changes to DCD Section 5.4.7 in accordance with NUREG-0800 Section 5.4.7, “Residual Heat Removal (RHR) System.” The changes are acceptable if the system continues to satisfy the relevant requirements in NUREG-0800 Section 5.4.7.

In Section 5.4.7.4.2, “Plant Cooldown,” an editorial change provides a description of the method used to control the cooldown rate as stated: “The cooldown rate is controlled by throttling the flow through the heat exchanger based on reactor coolant temperature.” Flow through the RNS heat exchangers will be controlled with the HX discharge valves, V006A(B). This is accomplished by switching the control signals on valves V008A(B) with those on valves V006A(B). The NRC staff agrees that this method of control will reduce the probability of an operator error during a plant cooldown condition because this is the standard method of controlling cooldown rate. Also, this revision is pictorially reflected on Figure 5.4-7 of the control signals to the valves. The NRC staff finds this change acceptable because it has the potential to improve operator performance by reducing the potential for operator error.

An additional revision to Figure 5.4-7 included a modification to V029, RNS motor-operated control valve downstream of the RNS HX connecting to the CVS. This change will allow the valve to auto close on high temperature. The reason for this change is to prevent high temperature fluid from passing through the demineralizers with the potential of damaging the resin. The resin in the CVS demineralizers can be damaged if the temperature through the demineralizers exceeds 60 °C (140 °F) for extended periods of time. The NRC staff finds this change acceptable because the change improves the reliability of the CVS system by reducing the potential damage to the resin.

In another revision to Figure 5.4-7, the RNS suction line relief valve is resized from a 10.2 cm (4 in) inlet and a 15.2 cm (6 in) outlet to a 7.6 cm (3 in) inlet and a 10.2 cm (4 in) outlet. RNS suction line relief valve resizing is implemented to reduce valve instability caused by the valve being fluid starved, which has the potential to cause valve chattering. The RNS relief valve is sized to mitigate primary overpressure events at low temperature conditions due to a heat injection and/or mass injection transient where the mismatch flow rate is no higher than 177 gpm. The staff finds this change acceptable because it improves the reliability of the valve by reducing potential damage to the valve due to unnecessary valve chatter.

The applicant modified DCD Section 5.4.7.1.2.1 to state that the CCW system supply temperature to the RNS HX is based on the maximum normal ambient wet bulb temperature as defined in Chapter 2, Table 2-1. The maximum normal ambient temperature is assumed for shutdown cooling. DCD Section 5.4.7.1.2.3 revises the maximum ambient design wet bulb temperature from 27.2 °C (81 °F) to 30.1 °C (86.1 °F) to facilitate consideration of other potential nuclear plant sites. The applicant has stated that the evaluation indicates there is sufficient margin in the systems design for the RNS, CCW system, and service water system (SWS) to maintain the same criteria and design basis with the increased ambient wet bulb temperature. In RAI-SRP5.4.7-SRSB-01, the staff requested that the applicant provide the results of the evaluation that demonstrate the RNS maintains sufficient margin. In its July 17, 2008, response, the applicant stated that "cooling the [in-containment refueling water storage tank] IRWST with the RNS, CCS, and SWS is not required to ensure that the plant can be maintained in a long-term safe condition." Therefore, under these circumstances, the staff agrees that the RNS, CCW system, and SWS designs would maintain sufficient margin with the revised higher ambient wet bulb temperature. The staff concludes that the proposed changes are acceptable because the analysis demonstrates that the RNS system can perform its intended function for normal operating conditions and is not required during transients since the RNS is not considered a safety-related system, i.e., no operational credit is assigned in Chapter 15, "Accident Analysis."

Section 5.4.7.2.1 was revised to clarify the temperature instrumentation employed to monitor the RCS hot leg for RNS operation during different RCS level conditions. For normal RCS inventory conditions, the RCS hot-leg wide-range temperature instruments are monitored; whereas, during reduced RCS inventory conditions, the RNS temperature instruments located upstream of the HX are monitored for the RCS hot-leg temperature. The staff finds that the revision is an editorial clarification with no impact on the RNS functionality. Therefore, the staff concludes the revision is acceptable.

Safety-related makeup water can be provided through the RNS for long-term post accident containment makeup. DCD Revision 17 changed Section 5.4.7.5 to state that this makeup is provided through the manual containment isolation test connection valve in the discharge of the RNS. This change simplifies the long-term makeup flow path setup by eliminating a component in the flow path, improves the RNS long-term makeup flow method, and has no impact on the functionality of the RNS. Therefore, the staff concludes that the change is acceptable.

In Tier 1, Table 2.3.6-1, the applicant deleted RNS HX A and B channel head drain valves RNS-PL-V046A and B and added RNS discharge containment isolation test connection RNS-PL-V012 to reflect the new RNS long-term makeup flow path to provide containment makeup. It also revised Figure 2.3.6-1 to reflect this change to the long-term makeup flow path configuration.

5.4.7.3 Conclusions

The staff reviewed the proposed changes to DCD Section 5.4.7 and concludes that the proposed changes improve overall system performance with the potential for improved operator performance without having a significant effect on the RNS design basis or RNS interfacing systems and meet the relevant acceptance criteria of NUREG-0800 Section 5.4.7.

6. ENGINEERED SAFETY FEATURES

Westinghouse has submitted information in support of its design certification (DC) amendment application that Westinghouse (the applicant) considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5), "Public inspections, exemptions, requests for withholding." The applicant has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears within "square brackets" as follows:

[ ]

The complete text of this chapter, including proprietary information can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

6.1.1 Metallic Materials

6.1.1.1 Summary of Technical Information

The applicant revised Section 6.1.1.2 of the AP1000 design control document (DCD) Tier 2 to reflect a proposed modification to DCD Tier 2, Table 5.2-2, "Reactor Coolant Water Chemistry Specifications," to allow the injection of zinc into reactor coolant water. The basis for this change appeared in Westinghouse Electric Company, LLC, technical report (TR)-32, APP-GW-GLN-002, "AP1000 Licensing Design Change Document Zinc Addition," dated April 5, 2006. The proposed change to Section 6.1.1.2 deletes the reference to zinc as a material that is not allowed to come into contact with engineered safety feature (ESF) components made of stainless steel. In Revision 17 of the DCD, the applicant changed Section 6.1.1.2 to make it consistent with Table 5.2-2.

The applicant revised DCD Tier 2, Section 6.1.1.3, to replace American Society for Testing and Materials (ASTM) A240 Type XM-29 with ASTM/American Society of Mechanical Engineers (ASME) A240/SA-240 UNS S32101 (LDX 2101) for the fabrication of the in-containment refueling water storage tank (IRWST). Westinghouse's TR-106, APP-GW-GLN-106, "AP1000 Standard Combined License Technical Report, AP1000, Licensing Design Changes for Mechanical System and Component Design Updates," Revision 1, dated September 28, 2007, identified and justified this change. Westinghouse's TR-134, "AP1000 DCD Impacts to Support COLA Standardization," Revision 5, dated June 27, 2008, details the modifications pertaining to DCD Section 6.1.1.3. DCD Tier 2, Revision 17, incorporates the proposed modifications.

In addition, the applicant revised DCD Tier 2, Section 6.1.1.4, to add a design requirement that aluminum excore detectors be enclosed in stainless steel or titanium to address, in part, Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance." The applicant also modified DCD Tier 1, Table 2.2.3-4, "Inspections, Tests, Analyses, and Acceptance Criteria," to require verification that the excore detector surfaces are made of stainless steel or titanium. TR-134 details the modifications to the DCD. The justification for these modifications appears in Westinghouse's TR-26, APP-GW-GLR-079, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Revision 7, dated February 26, 2010, and APP-GW-GLE-002, "Impacts to the AP1000 DCD to Address Generic Safety Issue (GSI)-191," Revision 7, dated July 13. DCD Tier 1 and 2, Revision 17, incorporates the proposed modifications.

6.1.1.2 Evaluation

DCD Tier 2, Section 6.1.1.2, states that lead, antimony, cadmium, indium, mercury, zinc, and tin metals and their alloys are not allowed to come into contact with ESF component parts made of stainless steel or high-alloy metals during fabrication or operation. The applicant proposed to modify DCD Tier 2, Section 6.1.1.2, to delete zinc as a material that is not allowed to come into contact with ESF components made of stainless steel. This proposed modification results from a proposed change in the reactor coolant water chemistry specifications, detailed in Table 5.2-2, to allow the injection of zinc into reactor coolant water. The basis for this proposed change relates to the benefits resulting from the addition of zinc to primary coolant which reduces radiation fields and the formation of crud. Section 5.2.3 of this safety evaluation report (SER) includes a detailed staff evaluation of the applicant's proposed change related to the addition of zinc to reactor coolant, which the staff finds acceptable. This proposed modification is generic and is expected to apply to all combined license (COL) applications referencing the AP1000 DC. Therefore, the proposed DCD change is acceptable pursuant to 10 CFR 52.63(a)(1)(vii), "Finality of standard design certifications," on the basis that it contributes to the increased standardization of the certification information.

DCD Tier 2, Section 6.1.1.3, describes the materials for nonpressure-retaining portions of ESFs in contact with borated water or other fluids. The IRWST liner and the passive containment cooling system (PCS) storage tank liner are primary examples of these items.

General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Basis," of Appendix A, "Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities," requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). In order for the IRWST to meet the requirements of GDC 4, the materials used must be compatible with ESF fluids.

Currently, AP1000 DCD, Section 6.1.1.3, states that ASTM A240 Type XM 29 (Nitronic 33) may be used to fabricate the IRWST. Table 6.1-1 specifies that XM-29 or TP304 will be used to fabricate the IRWST. The applicant proposed to modify the stainless steel surfaces of the containment internal structural modules as part of design change 049, which is described in TR-106. This design change will affect the IRWST, which is part of the ESFs. The proposed change to Section 6.1.1.3 specifies ASTM/ASME A240/SA240 UNS S32101, commonly known as LDX 2101, for fabrication of the IRWST.

The applicant provided its basis for the selection of LDX 2101 in TR-106, which stated that the proposed change to use LDX 2101 is because of the limited availability of Nitronic 33 in the required plate sizes. The applicant also stated that LDX 2101 has similar corrosion resistance and is easily welded.

LDX 2101 is an austenitic-ferritic (duplex) stainless steel. LDX 2101 provides adequate resistance to uniform corrosion, pitting, crevice corrosion, and stress-corrosion cracking in most media. It has higher mechanical strength than traditional stainless steels such as 304, 316L, and XM-29. LDX 2101 can be readily welded with commonly used welding processes. LDX 2101 is a relatively new material that was adopted in ASTM SA-240 in 2007. Its current commercial uses range widely from chemical storage tanks, waste water treatment facilities, and over-the-road chemical transportation tanks. This material is now widely used in several industries as a replacement for grade 304 and 316 stainless steels. The information provided by the applicant and the manufacturer's literature suggest that this material will perform adequately when exposed to borated water environments. Duplex stainless steels are considered to have more than adequate resistance to general corrosion and stress-corrosion cracking in aqueous solutions containing boric acid and chlorides.

A material's resistance to pitting can be compared to other materials using its pitting resistance equivalent (PRE) number. The PRE number is a theoretical method to compare the resistance to pitting and crevice corrosion of different types of stainless steels based on their chemical composition. In response to request for additional information (RAI)-TR106-CIB1-04, dated January 29, 2008, the applicant provided PRE numbers for stainless steel materials 304, 316L, and XM-29. The applicant also provided PRE numbers for LDX 2101, 2304 duplex stainless steel, and 2205 duplex stainless steel. The information provided by the applicant suggests that LDX 2101 has improved resistance to pitting and crevice corrosion when compared to 304, 316L, and XM-29 stainless steels. The applicant provided data in its January 29, 2008, letter that indicates that duplex stainless steels perform better than or equal to austenitic stainless steels, such as Types 304 and 316L, when exposed to borated water and chlorides. In addition, the applicant stated that it was just beginning its corrosion testing program for LDX 2101 base material and LDX 2101 welds. The staff notes that the information supplied by the applicant included data for duplex stainless steels 2304 and 2205 but did not include data for LDX 2101. While the staff agreed that LDX 2101 was likely to perform in a manner similar to 2304 and 2205 duplex stainless steels in borated water, confirmatory testing must be completed before the staff can approve the use of this material.

In supplemental RAI-TR106-CIB1-05, the staff asked the applicant to discuss its corrosion test plan and acceptance criteria for LDX 2101 base material and LDX 2101 weld filler materials. In addition, the staff requested that the applicant provide a technical justification for its testing plan and acceptance criteria that describe its adequacy to ensure that the materials will not be subject to general corrosion, stress-corrosion cracking, or other form of degradation from corrosion for the life of the plant.

The applicant responded to RAI-TR106-CIB-05 in a letter dated May 14, 2008. The applicant stated that it would conduct a confirmatory corrosion testing program to demonstrate the adequacy of LDX 2101. The applicant's confirmatory corrosion testing program includes LDX 2101 base material and weld filler materials that bound those filler materials that will be used during fabrication. Tests that will be conducted include uniform corrosion, stress-corrosion cracking, and crevice corrosion tests. The applicant stated that its test program was designed to establish test data on LDX 2101 material and its welds in terms of their susceptibility to degradation under exposure to oxygenated boric acid with halogen (chloride) contamination and in crevice corrosion conditions under accelerated service conditions to demonstrate a service life of 60 years. All tests will use Type 304 austenitic stainless steel as a reference sample. The staff notes that it approved Type 304 stainless steel, in addition to XM-29, for use in the fabrication of the IRWST liner in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004.

The staff found that the information that the applicant provided suggests that LDX 2101 duplex stainless steel material is likely to perform adequately in the IRWST environment. However, given the lack of specific corrosion data for LDX 2101, the staff asked, in RAI-SRP6.1.1-CIB1-02, that the applicant provide the results from its LDX 2101 corrosion testing program and describe the extent to which the results confirm that LDX 2101 will not be subject to general corrosion, stress corrosion, or other forms of degradation for the design life of the plant. The applicant responded in a letter dated June 30, 2010, and stated that it has completed a confirmatory testing program that included LDX 2101 and its associated welds under service conditions in the AP1000 IRWST. The test program included testing for three different types of corrosion: uniform corrosion, crevice corrosion, and stress-corrosion cracking. The testing also included Type 304 L stainless steel samples as reference samples for benchmarking. The staff notes that it approved Type 304 austenitic stainless steel for use in the IRWST, as documented in NUREG-1793. The applicant stated that the overall results of its confirmatory tests demonstrate that S32101 (LDX 2101) duplex stainless steel and its welds exhibited superior performance under accelerated service conditions in comparison with 304L stainless steel for use in the AP1000 structural modules [ ]. The staff notes that Types 304 and 304L are commonly used materials for applications such as the IRWST in operating plants. Westinghouse Commercial Atomic Power (WCAP)-17280-P, “Confirmatory Corrosion Testing of S32101/LDX2101 Duplex Stainless Steel Base and Weld Materials for the AP1000 Structural Module Applications,” issued June 2010, documents the test program and the results. The staff conducted an audit of WCAP-17280-P at Westinghouse’s Rockville, Maryland, office on June 28, 2010. As a result of the audit, the staff concluded that the test data in WCAP-17280-P provide reasonable assurance that LDX 2101 will not be subject to general corrosion, stress corrosion, or other forms of degradation for the design life of the plant and is, therefore, acceptable.

The staff finds that the proposed modifications to Section 6.1.1.3 are acceptable and meet the requirements of GDC 4 and the acceptance criteria of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 6.1.1, “Engineered Safety Features Materials,” Revision 2, issued March 2007. Revision 17 of the DCD and TR-134, Revision 5, incorporate the proposed changes as identified in TR-106. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

DCD Tier 2, Section 6.1.1.4, describes the materials’ compatibility with reactor coolant and ESF fluids. Section 6.1.1.4 currently states that in the postaccident environment, both aluminum and zinc surfaces in the containment are subject to chemical attack, resulting in the production of hydrogen. The applicant proposed to modify Section 6.1.1.4 to state that in the postaccident environment, both aluminum and zinc surfaces in the containment are subject to chemical attack, resulting in the production of hydrogen or chemical precipitants or both that can affect long-term core cooling. Primary sources of aluminum in the AP1000 containment are the excore detectors. To avoid sump water contact with the excore detectors, the applicant proposed to modify the DCD to state they will be enclosed in stainless steel or titanium housings. In addition, the applicant has proposed to modify DCD Tier 1, Table 2.2.3-4, to include inspections, tests, analyses, and acceptance criteria (ITAAC) to verify that exposed surfaces of the excore detectors are made of stainless steel or titanium. The applicant’s basis for the proposed change appears in Westinghouse DCD Impact Document and TR-26. These proposed modifications are intended to address, in part, GSI-191. Section 6.2.1.8 of this report provides a detailed staff evaluation of the applicant's design related to GSI-191. The staff finds the applicant's proposal to enclose the excore detectors in stainless steel or titanium acceptable because these materials will prevent the degradation of the aluminum excore detectors, which could result in chemical precipitants that can affect long-term core cooling. Revision 17 of the DCD incorporates the proposed changes, as identified above. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

6.1.1.3 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the AP1000 DCD, Section 6.1.1, are acceptable. Revision 17 to the AP1000 DCD incorporates the proposed changes, as identified in TR-134, TR-32, TR-26, TR-106 and the AP1000 DCD Impact Document. Furthermore, the staff finds that the conclusions in the above supporting documentation on the evaluation of these proposed DCD modifications are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

6.1.2 Organic Materials

6.1.2.1 Summary of Technical Information

The DCD changes described below are based on DCD Revision 17 and Westinghouse document APP-GW-GLE-002, Revision 7 (DCD Impact Document). The applicant modified DCD Tier 2, Section 6.1.2.1, in several places to change or clarify the types of coatings, the locations where they are used, and the associated quality assurance requirements.

The staff evaluated coatings for the certified AP1000 design based on NUREG-0800 Section 6.1.2. The staff concluded that the design met the quality assurance requirements of Appendix B to 10 CFR Part 50, as they relate to protective coatings. This conclusion was based on the design's conforming to the guidance in regulatory guide (RG) 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 1, including the provisions for design basis accident testing. RG 1.54, Revision 1, also provides guidance for coatings application and assessment.

The changes in DCD Revision 17 and the DCD Impact Document were proposed in order to meet the relevant requirements of GDC 35, "Emergency Core Cooling," GDC 38, "Containment Heat Removal," and 10 CFR 50.46(b)(5), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," for cooling water following a LOCA. Coatings are discussed in that context in Section 6.3 of the AP1000 DCD and 6.2 of the staff's corresponding safety evaluation. The proposed changes include the COL information required in Section 6.1.3.2 for consistency with Section 6.1.2. Additional information was provided in TR-26.

6.1.2.2 Evaluation

Inside containment, as described in the revised Section 6.1.2.1.5, inorganic zinc (IOZ) will be applied only to surfaces that exceed the temperature limit for epoxy during normal operation. All

IOZ inside containment will be classified as Service Level I. The staff finds these changes acceptable, since appropriately qualified epoxy coatings are suitable replacements for corrosion protection at lower temperatures, and because the Service Level I designation conforms to RG 1.54, Revision 1. Section 6.1.2.1.2 also introduces self-priming high-solids epoxy (SPHSE) without a zinc primer for structural modules, and the zinc/epoxy combination is eliminated except for the inside of the containment vessel up to seven feet above the operating floor. SPHSE is also added as a possible coating on concrete floors and walls. The staff finds this acceptable because the technical and quality requirements relating to the epoxy selection, and conformance to RG 1.54, Revision 1, are not changed. SPHSE is a standard type of coating, and SPHSE products have been design basis accident (DBA) qualified for steel and concrete. Eliminating the zinc/epoxy combination (except on the containment shell near the operating floor) is acceptable because there is no change in the quality requirements for the SPHSE.

Another significant change for inside containment is that coatings on manufactured components below the LOCA flood zone or in locations susceptible to debris transport must have a density of at least 1602 kilograms per cubic meter (kg/m^3) (100 pounds per cubic foot (lb/ft^3)) or a report, approved by the NRC, demonstrating the debris from that coating will not transport. This is discussed in DCD Sections 6.1.2.1.5 and 6.1.2.1.6. The staff finds this acceptable because the high density requirement limits transport of potential debris, while other coating requirements are not changed. SPHSE was also introduced as a coating for outside containment. According to the modifications in Section 6.1.2.1.4, carbon steel outside containment will be coated with either SPHSE or IOZ with epoxy top coat. Concrete floors and walls will be coated with epoxy or SPHSE. Section 6.1.2.1.6 was changed to clarify that procurement of Service Level II coatings outside containment, unlike inside containment, is not a safety-related, 10 CFR Part 50, Appendix B activity.

DCD Section 6.1.3.2, the COL information item called “Coating Program,” was modified to state that the COL applicant must include Service Level II coatings in the coatings program, and the coatings program includes inspection along with procurement, application, and monitoring. (This is also identified in the DCD as COL Information Item 6.1-2.) As indicated in Tier 2 Appendix 1A of the DCD, there is an exception to RG 1.54, Revision 1 due to the use of coatings inside containment that are designated Service Level II. The quality assurance requirements for these coatings are discussed in DCD Section 6.1.2.1.6. The change to Section 6.1.3.2 was proposed in a March 31, 2010, response to RAI-SRP6.1.2-CIB1-01. The staff found it acceptable because it makes the COL information required by Section 6.1.3.2 consistent with the information in Section 6.1.2.1.6. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Tier 1, Table 2.2.3-4, Item “x,” was modified to add two requirements to the existing requirement for nonsafety-related coatings used inside containment on walls, floors, ceilings, and structural steel (except for inside a chemical and volume control system (CVS) room that drains to the waste liquid processing system). The existing requirement (acceptance criterion) is a report concluding that these coatings have a dry film density of at least 1602 kg/m^3 (100 lb/ft^3). The two additional acceptance criteria are for a report showing that these coatings will not transport if the density is less than 1602 kg/m^3 (100 lb/ft^3) and for a report concluding that inorganic zinc coating used on these surfaces is safety Service Level I. (This ITAAC includes components in the LOCA flood zone, or above the flood zone and not in an enclosure.) The staff finds these changes acceptable because they supplement the existing ITAAC intended to ensure coatings chips generated by a LOCA will not transport.

6.1.2.3 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the AP1000 DCD, Section 6.1.2 are acceptable. The proposed changes are incorporated in Revision 17 of the DCD and in the AP1000 DCD impact document, APP-GW-GLE-002, Revision 7. Furthermore, the staff finds that conclusions about these DCD modifications in the documents cited above are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

6.2.1 Primary Containment Functional Design

6.2.1.1 Containment Pressure and Temperature Response to High-Energy Line Breaks

6.2.1.1.1 Wet Bulb Temperature

6.2.1.1.1.1 Summary of Technical Information

As described in APP-GW-GLE-036, “Impact of a Revision to the Current Wet Bulb Temperature Identified in Table 5.0-1 (Tier 1) and Table 2-1 (Sheet 1 of 3) of the DCD (Revision 16),” Revision 0, issued June 27, 2008, the site parameters for external wet bulb temperatures are increased from 26.7 °Celsius (C) to 30 °C (80 °Fahrenheit (F) to 86.1 °F) coincident and 29.7 °C to 30 °C (85.5 °F to 86.1 °F) noncoincident to encompass more sites in the eastern United States. The change to the coincident wet bulb temperature corresponds to an increase in relative humidity from 22 percent to 31 percent at the maximum dry bulb temperature of 46.1 °C (115 °F) and atmospheric conditions. The change to the noncoincident wet bulb temperature increases the temperature at which 100 percent relative humidity can occur. The external temperature and relative humidity are initial conditions in the containment analysis.

6.2.1.1.1.2 Evaluation

In response to RAI-SRP6.2.1.1-SPCV-05, dated October 1, 2008, the applicant stated that the containment analysis results are not sensitive to the external relative humidity. The staff audited one of the supporting analyses, Appendix A.3 to APP-GW-GSC-040, “AP1000 WGOTHIC Containment Models for Integrated Safety Analysis Evaluation: Disposition of Design Change Proposals and Modification of the Containment Model,” and observed that when the relative humidity of the NRC-approved AP1000 WGOTHIC model for a double-ended cold-leg guillotine (DECLG) LOCA was increased from 22 percent to 31 percent at an external temperature of 46.1 °C (115 °F), the resulting peak pressure increase was negligible. This study also demonstrated that the increase to noncoincident wet bulb temperature is less limiting than the increase to coincident wet bulb temperature with respect to peak pressures. Additional sensitivity studies documented in Section 5.6 of WCAP-15846, “WGOTHIC Application to AP600 and AP1000,” Revision 1, issued March 2004, showed that the AP600 WGOTHIC LOCA and main steamline break models are insensitive to external humidity for select initial conditions. The staff ran confirmatory analyses using the CONTAIN model of the AP1000 containment developed by the staff during the DCD review. When the outside relative humidity in this model was increased to 31 percent, there was negligible impact on the peak pressures resulting from a main steamline break and DECLG. There was also a negligible increase in the pressure 24 hours after the DECLG LOCA.

While the staff found the increase to external wet bulb temperatures acceptable, it was not clear how the containment analyses referenced in the DCD would incorporate this change. In its July 12, 2009, Revision 2 response to RAI-SRP6.2.1.1-SPCV-06, the applicant stated that DCD changes to reference a more recent WGOTHIC model, which includes the increase to external wet bulb temperature, will be included in its submittal of APP-GW-GLR-096, “Evaluation of the Effect of AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis.” Revision 1 of APP-GW-GLR-096 was issued August 2010, and it does propose adding APP-GW-GLR-096 as a reference to the DCD. The staff confirmed that the model used in this study includes the increased wet bulb temperature, and is satisfied with the response. The evaluation of APP-GW-GLR-096 is documented in Chapter 23.

6.2.1.1.1.3 Conclusion

Based on the evaluations by the applicant and the confirmatory analysis by the staff, the staff concludes that that the containment functional design capability is essentially unchanged by the proposed increase in maximum site wet bulb temperatures. The conclusions reached in NUREG-1793, Sections 6.2.1 and 6.2.1.1, remain applicable, including that the design is compliant with regulatory requirements.

6.2.1.1.2 External Pressure Analysis

6.2.1.1.2.1 Summary of Technical Information

In DCD Revision 15, the maximum external pressure event is alleviated by the operator action of opening either set of purge valves. On August 16, 2010, the applicant submitted Change Number 74 to add a vacuum relief system that replaces the operator action. The “NRC Review Package” attached to this letter includes the revised analysis and proposed DCD changes for Section 6.2.1.1.4.

6.2.1.1.2.2 Evaluation

The staff evaluation of the adequacy of the vacuum relief system to mitigate the maximum expected external pressure scenario as described in Change Number 74 will be included in Chapter 23.

6.2.1.1.2.3 Conclusion

The staff’s conclusion regarding the external pressure analysis is provided in Chapter 23.

6.2.1.2 Subcompartment Analysis

6.2.1.2.1 Summary of Technical Information

APP-GW-GLR-016, “AP1000 Pressurizer Design,” issued May 2006, describes changes made to the diameter and height of the pressurizer and to the wall heights of the pressurizer compartment in order to obtain satisfactory piping analysis results. As a result of the shorter pressurizer walls, the applicant decreased the upper elevation of the assumed pressurizer spray break in DCD Tier 2, Section 6.2.1.2.3.2, from 52.1 to 49.7 meters (m) (171 to 163 feet (ft)).

6.2.1.2.2 Evaluation

The subcompartment analysis previously approved by the staff used the TMD computer code with models described in WCAP-15965, "AP1000 Subcompartment Models," issued November 2002. As reported in the July 18, 2008, response to RAI-SRP6.2.1.2-SPCV-01, the applicant performed a conservative calculation to evaluate the impact of the pressurizer changes. The results showed that the differential pressure (dP) in the pressurizer cubicle remained below the 34.5 kilopascal (kPa) (5 pounds per square inch differential (psid)) structural threshold. The staff ran a confirmatory analysis using COMPARE with input compiled from the pressurizer compartment model described in WCAP-15965 and the changes to the pressurizer described in APP-GW-GLR-016. The staff analysis predicted that the pressurizer changes would increase the maximum dP from [ ], which is consistent with the applicant's results.

The revised height of the pressurizer wall is 48.8 m (160 ft). Based on the July 18, 2009, response to RAI-SRP6.2.1.2-SPCV-02, the pressurizer spray line extends 0.9 m (3 ft) above the top of the wall; therefore, changing the upper elevation for the assumed pressurizer spray break to 49.7 m (163 ft) is appropriate.

While the staff found the changes to the pressurizer and pressurizer compartment acceptable with respect to subcompartment analyses, it was not clear how the revised analysis would be incorporated into the DCD. In a letter dated August 31, 2009, in response to RAI-SRP6.2.1.2-SPCV-01 Revision 2, the applicant revised the DCD to state that the impact of the dimensional changes to the pressurizer and pressurizer compartment on the subcompartment analysis was evaluated in APP-GW-GLR-138, "Evaluation of the Pressurizer Changes on the AP1000 TMD Analyses," issued August 2009, and the existing conclusions remain valid. The staff is satisfied with this response. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

6.2.1.2.3 Conclusion

Based on the results of the applicant's analysis and the staff's confirmatory calculations, the staff agrees that the changes made in the DCD related to the pressurizer compartment have a negligible impact on the AP1000 subcompartment analysis. The conclusions reached in NUREG-1793, Section 6.2.1.2, remain applicable, including that the containment subcompartment pressurization analysis is acceptable.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

6.2.1.3.1 Summary of Technical Information

While the applicant did not change DCD Section 6.2.1.3.2.1 regarding the 1 percent full-power allowance for calorimetric error in the energy release calculations, the applicant's response to RAI-SRP15.0-SRSB-02 proposes a change to this section as discussed below.

6.2.1.3.2 Evaluation

For analyses of heat sources during a postulated LOCA, paragraph I.A of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 permits an assumed power level allowance of less than 2 percent full power if it has been demonstrated to account for uncertainties related to power level instrumentation error. The power level used to determine the maximum containment pressure in Chapter 6 of the DCD was based on a calorimetric error of less than 2 percent full power, but there was no associated justification. In a May 6, 2009, response to RAI-SRP15.0-SRSB-02, the applicant stated that the AP1000 will use the proven technology of high-accuracy instrumentation to demonstrate a 1 percent design uncertainty and added COL Information Item 15.0-1, described in DCD Section 15.0.15, to track this commitment. The applicant also added a note to the DCD Section 6.2.1.3.2.1 assumptions on energy release to reference this COL item. This is appropriate as it provides justification for the uncertainty value, as required by the stated criteria.

6.2.1.3.3 Conclusion

The change is acceptable, and the conclusions reached in NUREG-1793, Sections 6.2.1.3 and 6.2.1.4, remain applicable, including that the methods and assumptions are acceptable for the licensing analyses.

6.2.1.8 Adequacy of In-Containment Refueling Water Storage Tank and Containment Recirculation Screen Performance

DCD Tier 2, Section 6.3.2.2.7, describes the evaluation of the water sources for long-term recirculation cooling following a LOCA, including the design and operation of the AP1000 passive core cooling system (PXS) debris screens. DCD Section 6.3.8 describes the associated COL information items, and DCD Tier 1, Section 2.2.3, includes the associated design descriptions and ITAAC. DCD Revision 17 incorporated many changes to these sections, as did APP-GW-GLE-002, which identified changes beyond those included in DCD Revision 17. Because the revisions are so extensive, the staff will not address each change independently but will perform a complete evaluation of the final configuration. As such, this section of the SER amendment replaces the analysis documented in NUREG-1793, Revision 0, in its entirety. In a subsequent revision to the AP1000 DCD, the applicant incorporated the DCD text proposed in APP-GW-GLE-002, Revision 7, dated July 13, 2010.

One of the changes made is the closure of COL Information Item 6.3-2 from DCD Tier 2, Table 1.8-2. DCD Tier 2, Section 6.3.8.2, “Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA,” originally made the following commitment:

> The Combined License applicants referencing the AP1000 will perform an evaluation consistent with RG 1.82, Revision 3, and subsequently approved NRC guidance, to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA. As discussed in DCD Subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design. The evaluation will consider resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris will be based on sample measurements from operating plants. The evaluation will also consider the potential for the generation of chemical debris (precipitants). The potential to generate such debris will be determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

In DCD Revision 17, the applicant stated the following:

> The Combined License information item requested in this subsection has been fully addressed in APP-GW-GLR-079 (Reference 3), and the applicable changes are incorporated into the DCD. The design of the recirculation screens is complete. Testing to assess the screen performance and downstream effects is complete. A study of the effects of screen design and performance on long-term cooling is complete. No additional work is required by the Combined License applicant to address the aspects of the Combined License information requested in this subsection.

The following Commission regulations are related to the evaluation of the water sources for long-term recirculation cooling following a LOCA:

- GDC 35, as it relates to providing abundant emergency core cooling to transfer heat from the reactor core following a LOCA

- GDC 38, as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain these indicators at acceptably low levels

- 10 CFR 50.46(b)(5), as it relates to requirements for long-term cooling in the presence of LOCA-generated and latent debris

As directed by NUREG-0800 Section 6.2.2, "Containment Heat Removal Systems," Revision 5, the staff performed the review in accordance with RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, issued November 2003, as supplemented by the Nuclear Energy Institute (NEI) Guidance Report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, Volume 1, issued December 2004, and the associated NRC safety evaluation, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," issued December 2004. The review was also informed by WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, dated August 2007, as supplemented by "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16406-P," "Evaluation of Downstream Sump Debris Effects in Support of GSI-191, Revision 1," Revision 0, dated December 20, 2007; the NRC letter dated March 28, 2008, "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," with enclosures addressing the areas of chemical effects, coatings, and head loss testing; the final safety evaluation by the Office of Nuclear Reactor Regulation on TR WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated December 21, 2007; and the NRC letter dated April 6, 2010, "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02."

In addition to the DCD, as modified by APP-GW-GLE-002, the staff reviewed APP-GW-GLR-079-P and APP-GW-GLR-086-NP (TR-26), "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Revision 8, dated July 20, 2010, and APP-GW-GLN-147-P and -NP (TR-147), "AP1000 Containment Recirculation and IRWST Screen Design," Revision 3, issued November 2009, which are identified as DCD references. The following submittals from the applicant provided additional information: WCAP-16914-P and -NP, "Evaluation of Debris Loading Head Loss Tests for AP1000 Recirculation Screens and In-Containment Refueling Water Storage Tank Screens," Revision 5, issued June 2010; WCAP-17028-P and -NP, “Evaluation of Debris Loading Head Loss Experiments Across AP1000 Fuel Assemblies During Post-Accident Recirculation,” Revision 6, issued June 2010; APP-PXS-GLR-001, “Impact on AP1000 Post-LOCA Long-Term Cooling of Postulated Containment Sump Debris,” Revision 4, dated February 26, 2010; APP-GW-GLR-092-P and APP-GW-GLR-093-NP, ”Statistical Evaluation of AP1000 Fuel Assembly Debris-Loading Head loss Tests,” Revision 0, issued February 2010; and APP-GW-GLR-110-P and APP-GW-GLR-111-NP, “Boric Acid Precipitation Tests During Post-LOCA Conditions,” Revision 0, issued February 2010.

The applicant responded to staff RAIs in letters dated November 6 and 11, 2008; April 22, May 12, May 13, May 14, May 15, May 20, May 27, June 4, July 7, July 22, July 31, September 17, September 22, and November 2, 2009; and January 29, February 26, March 12, March 26, April 1, April 16, April 26, April 29, May 11, May 13, May 28, June 14, June 28, June 30 and July 30, 2010.

6.2.1.8.1 Summary of Technical Information

The AP1000 has two containment recirculation screens and three IRWST screens to capture debris following a LOCA. All screens are designed to be vertically oriented and have solid top covers so that debris that settles out of the water does not fall on the screening surfaces. The screens are constructed of stainless steel to be corrosion resistant and comprise individual pockets to provide greater filtering areas for a given volume. All AP1000 screens use the same pocket geometry, with each pocket having a frontal face area greater than or equal to 40 square centimeters (cm^2) (6.2 square inches (in^2)), a screen surface area greater than or equal to 903 cm^2 (140 in^2), and a hole size less than or equal to 1.59 millimeters (mm) (0.0625 inch (in)). This pocket design also provides the trash rack function because it prevents a single object from blocking a large portion of the screen.

Three separate screens are located inside the IRWST at the bottom of the tank. Two of the screens are located at opposite ends of the tank, each with a frontal area greater than 1.9 square meters (m^2) (20 square feet (ft^2)) and a surface area greater than 46.5 m^2 (500 ft^2). The third screen, with a frontal area greater than 3.7 m^2 (40 ft^2) and a surface area greater than 92.9 m^2 (1,000 ft^2), is located in the center of the tank. Each of the smaller screens supplies one of the two recirculation lines and is joined to the center screen through cross-connect piping, which distributes the flow. The lowest screening surfaces are located 0.2 m (6 in) above the IRWST floor to prevent debris from being swept along the floor into the screen. During recirculation, the only flow entering the IRWST is the steam condensate that forms inside of the containment shell and collects in the IRWST gutter. The gutter feeds the IRWST through two drainpipes that terminate within 1.5 m (5 ft) of the IRWST floor and at least 3.7 m (12 ft) away from any screen face, which prevents debris from entering the tank close to the screens. For minimum floodup conditions during recirculation, the water level is a few inches above the top of the screen assembly.

The containment recirculation sump for the AP1000 is the loop compartment. Two containment recirculation screens, each with a frontal area greater than 9.8 m^2 (105 ft^2) and a surface area greater than 232 m^2 (2,500 ft^2), are located next to one another along walls on the loop compartment floor. The loop compartment floor is 3.5 m (11.5 ft) above the reactor vessel cavity floor, which is the lowest level in containment. Each screen supplies one of the two recirculation lines. The screens are cross-connected with flow channels so that even if only one recirculation line is operating, both containment recirculation screens will be available to filter the flow. A 0.6 m (2 ft) high curb is located in front of the screens to prevent debris from being swept along the floor into the screen. Protective plates, which cover the screen and extend outward at least 3 m (10 ft) in front of the screen face and 2.13 m (7 ft) beyond the sides of the screen, are located no more than 0.3 m (1 ft) above the top of the containment recirculation screens. These plates are designed to prevent debris from settling into the water close to the screens. During recirculation, even at minimum floodup conditions, the water level remains about 3 m (10 ft) above the top of the screens.

The AP1000 is equipped with an active nonsafety-related injection and recirculation system (the residual heat removal system or RNS). This system is an investment protection system and will be used following a LOCA if there is power and the system is available. The RNS initially injects water from the cask loading pit and then switches to recirculate the sump water. The system is not credited in the safety analysis; however, the applicant has evaluated the screens and core assuming the higher RNS flow rates to demonstrate that the RNS system will function following a LOCA.

During the AP1000 recirculation phase, some portions of the reactor coolant system (RCS) piping are submerged. LOCAs resulting from pipe breaks in these locations will then be flooded, allowing sump fluid to bypass the screens and flow directly to the core. Because of this, the core is evaluated as a separate debris filtering location.

6.2.1.8.2 Evaluation

6.2.1.8.2.1 Break Selection

A primary objective of break selection is to identify the break location that results in debris generation that produces the largest head loss across the screens. Section 3.3 of NEI 04-07 describes a process; whereby, different break locations are systematically evaluated to determine which is limiting. In the AP1000 design, there are only three sources of debris that transport with the recirculating water: latent or resident containment debris, debris from postaccident chemical effects, and debris from coatings located in the zone of influence (ZOI) of a LOCA jet. The limiting break for each filtering location (screens or core) is determined by examining the debris generation and transport assumptions for each of these debris types.

6.2.1.8.2.1.1 Containment Recirculation Screens and In-Containment Refueling Water Storage Tank Screens Break Selection

In the AP1000, the amounts of latent and chemical debris generated and transported to the containment recirculation and IRWST screens are independent of break location; therefore, the only consideration for break selection is coatings debris. Because all coatings in the ZOI are assumed to generate fine transportable particulates, the limiting break location is the one whose ZOI contains the largest amount of coatings. The applicant described its break selection process for cold leg (CL) and hot leg (HL) LOCAs in the June 30, 2010, Revision 2 response to RAI-SRP6.2.2-SPCV-25. The applicant determined that the limiting breaks would occur in the largest diameter lines: the main loop CL, which has a 55.9 centimeter (cm) (22 in) inner diameter (ID) and the main loop HL, which has a 78.7 cm (31 in) ID. These breaks encompass potential breaks on smaller lines because the main loop pipes are located in the same general area and have significantly larger ZOIs than the smaller lines.

The applicant then selected the terminal ends of the main loop lines as potential break locations and quantified the amount of both epoxy and inorganic zinc coatings at each site by multiplying the surface area of coated beams, pipes and flat surfaces located in the ZOI by the coating thickness and density. The limiting CL location was the terminal end at reactor coolant pump (RCP) #2, in the west loop compartment. The limiting HL break was the terminal end at either steam generator, due to similarity between the compartments. The applicant then assessed potential break locations at 5 ft intervals along each CL and HL line, as recommended by the safety evaluation on NEI 04-07, to demonstrate that these sites, with potentially different coating inventories were bounded by the limiting breaks. This analysis employed plant layouts to identify the location of coated surfaces such as walls, plates, pipes and beams with respect to each postulated ZOI. When necessary, the applicant estimated the increase or decrease to each type of coated surface to show that the net amount of coatings was bounded by the limiting break.

The staff finds that the spectrum of breaks evaluated is acceptable because, while the unique characteristics of the AP1000 greatly simplify the break selection process, the applicant's procedure meets the intent of the NEI 04-07 and the related safety evaluation, and Regulatory Position C.1.3.2.3 of RG 1.82, Revision 3.

6.2.1.8.2.1.2 Core Break Selection

Because of the relatively high containment floodup level during long-term recirculation operation in the AP1000 design, some LOCA break locations will be flooded, resulting in a portion of recirculation flow with unfiltered containment debris entering the reactor core through the submerged break. The DECL break at the reactor vessel, DEHL break at the reactor vessel, and double-ended direct vessel injection (DEDVI) line break in the loop compartment are breaks that could be submerged in the flooded containment sump and allow unfiltered debris into the reactor vessel during the post-LOCA long-term cooling phase.

The objective of break selection is to choose a limiting LOCA break location for the GSI-191 long-term cooling evaluation. The break selection considers two aspects: (1) the limiting break, which contributes the most unfiltered debris to the core region causing core flow blockage; and (2) the limiting break for long-term core cooling evaluation, which results in the worst core heatup because of early containment recirculation initiation at a higher decay heat level.

In Section 6.2.1.8.2.6 of this report, the staff discusses the containment debris transport for the DECL break, DEDVI line break, and DEHL break and the evaluation of the percentage of the containment debris entering the reactor vessel for these breaks. The percentage of the debris that might be transported into the reactor vessel without screening by the containment recirculation screens is determined by integrating the relative recirculation flows through the break and through the intact DVI lines of the PXS. Flow split calculation for the DECL break identified that [ ] percent of the water in the containment will come through the DECL break, compared to the DEDVI line break with [ ] percent flow split. Therefore, the applicant determined that the DECL break at the reactor vessel is the limiting break with respect to debris transport to the core, with a flow split conservatively rounded up to 90 percent. The applicant assumed that all of the latent debris in the containment would be in the containment water and, therefore, 90 percent of containment debris would enter the reactor vessel through the submerged DECL break during the post-LOCA recirculation long-term core cooling phase, bypassing the IRWST and containment recirculation screens. This 90 percent debris bypass calculated with a DECL break is the design-basis value used to determine the containment debris bypass to the core for the AP1000 design.

For its long-term cooling evaluations performed in the AP1000 DCD, Revision 17, the applicant selected the DEDVI line break in the PXS room as the limiting long-term cooling case because it minimized the cooling water injection head at the highest decay heat generation rate. The case analyzed in the DCD is the continuation of a small-break LOCA. The DCD long-term cooling evaluations used the PXS valve room as the break location since natural circulation flow losses from the break to the core inlet are greatest, thereby minimizing the amount of flow into the core inlet from the break. The DCD case did not consider the GSI-191 concern that evaluates debris entering the core through the break location. Since the DEDVI break in the PXS room has only a small amount of debris accessible to the break (i.e., only debris in the PXS room floodup water volume can enter the break location), the applicant considered the DEDVI line break in the loop compartment as the limiting break for unfiltered debris entry into the core.

The applicant performed long-term cooling sensitivity studies, described in APP-PXS-GLR-001, Revision 4. The objective of these sensitivity studies was to demonstrate that core cooling margins are maintained when large, arbitrary, nonmechanistic head losses are added to the containment recirculation screens, IRWST screens, and the core inlet. Section 6.2.1.8.2.7 of this report includes additional discussion of these sensitivity studies. As discussed in Section 4 of TR-26, the applicant selected the direct vessel injection (DVI) break for the long-term cooling sensitivity analyses with debris-induced core head loss for the following reasons:

- A DECL break has significantly less resistance to flow than the DVI break; therefore, the DVI break will minimize the water flow to the core. For this reason, only the DVI breaks were analyzed with WCOBRA/TRAC in APP-PXS-GLR-001, Revision 4.

- The lower elevation of the DEDVI break also allows water from the containment to flow into the RCS through the break sooner than for a DECL LOCA; therefore, the debris will start to accumulate sooner with a higher decay heat.

- A DEDVI break results in higher decay heat levels at the time recirculation begins compared to a DECL LOCA. With a DEDVI break, a portion of the IRWST injection flow will spill out of the break into containment, thereby reducing the time of IRWST injection.

In the post-LOCA long-term cooling sensitivity studies, described in APP-PXS-GLR-001, Revision 4, the DEDVI line break in a PXS room and in the loop compartment adjacent to the DVI inlet nozzle, respectively, were assumed from the analysis. In Cases 1 through 5 and Case 7, which have smaller assumed core inlet flow resistances, the DVI line break was assumed in the PXS room, where the floodup level is lower than the floodup level in the loop compartment and is, therefore, more conservative. In other cases with larger core inlet flow resistances simulating a larger amount of debris entering the core, the DVI break is assumed to be in the loop compartment. This is a more realistic assumption because a DVI break in the loop compartment would be exposed to all of the latent debris in the containment to the break location; whereas, a DVI break in the PXS room would only expose the break to the small amount of debris that is located in the PXS room. Therefore, significantly more debris is available to enter the DVI break in the loop compartment. The sensitivity study Case 10, which is a DVI break in the loop compartment with significant flow blockage at the core inlet, was determined to be the limiting break for water flow and debris to the core.

As discussed in Section 6.2.1.8.2.6 of this report, the HL break was determined not to be a significant deterrent for long-term cooling. Based on fuel assembly testing, the debris that enters the top of the core will be broken up by the two-phase flow leaving the core, and core cooling is maintained by the intact DVI flow.

On the basis of its review, the staff finds that the applicant's assumption that 90 percent of the total latent debris in the AP1000 containment can enter the core inlet through the DECL break is conservative, and that the applicant's determination that the DECL break is the design-basis break location for debris transport to the core and for potential debris blockage of the core inlet is acceptable. Additionally, the staff finds that the applicant's use of the DEDVI break in the loop compartment for the long-term core cooling sensitivity studies while using the 90 percent debris bypass obtained from the DECL break to determine the debris-induced core entrance head loss is conservative and acceptable.

6.2.1.8.2.2 Zone of Influence/Debris Generation and Characterization (Excluding Coatings)

The ZOI is the volume about the break in which the LOCA break jet forces would be sufficient to damage materials. Debris generation is the amount of debris generated by these forces, and debris characterization establishes the properties of this debris. The staff considered Section 3.4 of NEI 04-07 and the related safety evaluation when evaluating this section, which addresses all debris types except coatings, which are discussed in the following section.

In the AP1000, metal reflective insulation (MRI) or a suitable equivalent is specifically required on the reactor vessel, RCPs, steam generators, pressurizer, and all ASME Code Class 1 lines. MRI is also required at any location within the insulation ZOI, which is defined in DCD Tier 2, Section 6.3.2.2.7.1, for two situations:

(1) When there are intervening components, supports, structures, or other objects, the ZOI includes the spherical region within a distance equal to 29 IDs of the pipe break for Min-K, Koolphen-K or rigid cellular glass or 20 IDs of the pipe break for other types of insulation.

(2) When there are no intervening components, supports, structures, or other objects, the ZOI is a cylindrical volume extending out from the break a distance of 45 IDs along an axis that is a continuation of the pipe axis and a distance of 5 IDs radial to the pipe axis.

The first ZOI definition is consistent with Section 3.4 of the safety evaluation on NEI 04-07, which considers the spherical ZOI a practical approximation of the jet impingement damage zone and provides appropriate radial values for spherical ZOI for specific insulation types in SER Table 3-2. The spherical radii used to describe the AP1000 ZOI are bounded by the values in this table, except for rigid cellular glass, which NEI 04-07 and the safety evaluation do not address. Since no data are available for rigid cellular glass, the applicant used the maximum ZOI of 29 IDs from Table 3-2 of the safety evaluation on NEI 04-07. The staff accepts this approach, which is further supported by the statements in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," issued February 2003, that this insulation will float indefinitely even if damaged. The second ZOI definition is identical to that certified in Revision 15 of the DCD. It was evaluated and found to be acceptable in NUREG-1793, Revision 0, based on existing tests and analysis.

If insulation in the AP1000 ZOI is not MRI, it must meet the DCD definition of suitable equivalence, which requires that the insulation be tested at conditions that bound the AP1000 operation and that if debris is generated it must not transported to any of the AP1000 filtering locations. It also requires that the NRC approve the test applicability and any subsequent analysis. This is appropriate because there are no clearly defined protocols for jet impingement testing, and all previous submittals on this type of testing were subject to staff evaluation.

DCD Tier 2, Section 6.3.2.2.7.1, Item 10 prohibits other potential sources of fibrous material, such as ventilation filters or fiber producing fire barrier, in the insulation ZOI. The staff agrees that this design commitment, in combination with the previously discussed insulation commitments, excludes all potential sources of fibrous debris except latent debris from the ZOI.

In RAI-SRP6.2.2-CIB1-28 (and supplements), the staff asked the applicant to clarify how concrete in containment is treated as a debris source, including chemical, coatings, and particulate debris. In letters dated January 10, 2010 and April 26, 2010, and July 30, 2010, the applicant explained that coatings are assumed to fail as particles within the ZOI and transport, and as chips outside the ZOI and not transport. In addition, the applicant explained that all concrete surfaces flooded following a LOCA are assumed to react with the water pool and potentially contribute to chemical debris. The staff found these assumptions acceptable, as discussed in Sections 6.2.1.8.2.3 and 6.2.1.8.2.4 of this report.

With respect to LOCA-generated particulate debris, the applicant explained that concrete is not a likely source of particulate debris for the AP1000 because the only concrete surfaces inside containment impacted by LOCA jets are the concrete floors. The containment walls and ceilings are constructed using steel-lined concrete modules. For AP1000, concrete debris is not considered in the composition of particulate debris based on the proximity of the concrete surfaces to the potential break locations, the orientation of the concrete with respect to the break, and the sharp decrease in the pressure of a LOCA jet as a function of distance.

In the July 30, 2010, letter, the applicant addressed the staff's question regarding the concrete damage apparent in the German HDR test results documented in NUREG/CR-0897, "Steam-Water Mixing and System Hydrodynamics Program, Quarterly Progress Report, Jan.-Mar. 1979". The applicant questioned the applicability of these results, since there is limited information about the testing and multiple tests led to the damage shown in NUREG/CR-0897. The applicant referenced WCAP-7391, "Pressurized Water and Steam Jet Effects on Concrete," as a more relevant study. This report documents results of jet impingement tests that were performed in 1970 by directing a [ ] pounds per square inch gauge (psig), water jet at [ ] concrete slabs. The July 30, 2010, letter also provides jet impingement pressures calculated for double-ended breaks in the limiting case of a pipe parallel to the AP1000 floor. As discussed below, the applicant performed further analysis using the WCAP-7391 test results and jet impingement calculations.

Although Section 1.3.2.4 of RG 1.82, Revision 3, states that erosion of concrete should be considered a potential source of particulate debris, the staff has not developed guidance for a ZOI or quantification of the debris. The applicant assessed the effect of potential concrete erosion debris on the available debris margins in the design basis. In its analysis, the applicant derived a threshold destruction pressure based on the WCAP-7391 jet impingement testing of concrete beams, and then determined whether any double-ended pipe breaks or longitudinal breaks ("split or side breaks") could exceed the threshold pressure. According to WCAP-7391, the tests did not erode the concrete itself, [ ]. In those tests, the minimum value of L/D (the length to diameter ratio which is the distance from the break to the target divided by the pipe inside diameter) was [ ] for the WCAP-7391 test conditions (CL conditions for the AP1000). The L/D ratio is a parameter used to characterize impingement pressure as a function of distance from a jet nozzle. Using the American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," jet model to calculate pressure contours for these conditions, the applicant calculated a stagnation pressure of [ ] at the concrete surface. For AP1000 HL conditions and a stagnation pressure of [ ], the corresponding L/D was [ ]. The applicant concluded that these tests demonstrated that jets from double-ended breaks in the AP1000 would not damage concrete at L/D values equal to or greater than [ ] for the HL and [ ] for the CL.

Therefore, for this margin assessment, the applicant set the acceptance criteria for concrete damage at [ ] and applied the same jet model to the double-ended break locations. From these calculations the applicant determined that no locations exceeded the [ ] criterion at the concrete surface, regardless of whether the pipe was oriented perpendicular or parallel to the floor. The staff found this approach acceptable because the WCAP-7391 tests showed no concrete damage and the staff has accepted the ANSI/ANS model as a basis for calculating pressure contours from LOCA jets. The staff evaluated the ANSI/ANS model in the December 2004, safety evaluation of NEI 04-07.

Continuing with the margin assessment, the applicant also evaluated the potential for concrete damage from longitudinal breaks (also referred to as “split breaks” or “side breaks”) at the same locations (module floors). For the geometry of the break, the applicant used one-half of the pipe inside diameter as the width and two times the inside diameter as the length. The applicant used this approach to conform to Branch Technical Position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.” The applicant then used the ANSI/ANS 58.2-1988 jet model to determine the distance to where the impingement pressure is less than the [ ] acceptance criterion for concrete damage. Based on these calculated pressures for longitudinal breaks, the applicant identified five lines for which the calculated jet impingement pressure on concrete exceeds the [ ] acceptance criterion: [ ].

To estimate the amount of concrete debris generated by these breaks, the applicant assumed that all of the concrete becomes debris where the jet impingement pressure exceeds [ ]. [ ]. The applicant also stated that the ellipsoid shape would be partially filled with steel reinforcement bar rather than concrete, but in this analysis the ellipsoid was assumed to be all concrete.

The applicant then assumed [ ]. The applicant considered this a conservative size distribution relative to the appearance of the concrete damage in photographs of the German HDR test results documented in NUREG/CR-0897. The applicant did not attempt to estimate the size of fine debris in these photographs, but concluded that the concrete debris was mostly in the form of large pieces several inches in length and width. Given the lack of experience with concrete debris generation from impingement, the wide size range of the materials in concrete, and the appearance of the debris generated in the HDR test, the staff found that this size distribution is a reasonable assumption for the concrete debris. The entire concrete debris surface area was assumed to react chemically with the sump fluid.

By summing the concrete, coatings, and latent particle mass at each break location, the applicant concluded that all locations were bounded by the particulate quantities used in the screen and fuel assembly tests. With respect to chemical debris, all of the concrete particulate was assumed to react with the water, potentially generating additional sodium aluminum silicate and calcium phosphate. The applicant determined that the concrete particulate debris added approximately [ ] to the [ ] of concrete already used in chemical debris calculation, for a total of [ ]. (As indicated in Section 6.2.1.8.2.4 of the SER, all concrete in the post-LOCA flood-up zone was assumed, for the design basis analysis, to have the coating removed and, therefore, assumed to react chemically with the sump fluid.) The total calculated amount of chemical debris, including that from the concrete particulate, did not exceed 25.9 kilogram (kg) (57 pounds (lb)), which is the design basis amount of chemical debris that was scaled for use in the screen and fuel assembly testing. The staff performed confirmatory calculations and reached the same conclusion regarding the surface area of the concrete debris and the increase in chemical debris.

Based on this analysis the applicant concluded the following from the margin assessment: (1) the only concrete surfaces potentially exposed to impingement from a LOCA jet are limited floor areas of modules; (2) double-ended breaks of the pipes nearest the concrete do not generate concrete debris, regardless of orientation (perpendicular or parallel); (3) split or side breaks in the pipes nearest the concrete may generate concrete debris from erosion; and, (4) the estimated quantities of concrete particulate and associated chemical reaction debris are bounded by the screen and fuel assembly head loss tests that support the proposed licensing basis.

The staff found the applicant's evaluation acceptable to address Section 1.3.2.4 of RG 1.82, Revision 3, because the evaluation supports the position that concrete debris from jet impingement for the AP1000 plant is unlikely, and because a reasonable estimate of the amount of particulate and chemical debris that could be generated from concrete by a LOCA is within the AP1000 design basis.

Debris Characterization

The next step in the evaluation is to characterize the generated debris for input to the transport analysis. The applicant assumed that MRI in the ZOI degrades to pieces of crumpled foil as small as 1.3 cm by 1.3 cm (0.5 in by 0.5 in), which is consistent with the smallest classification of blast-tested MRI in NUREG/CR-6808. The applicant did not provide a quantity of degraded MRI because the subsequent transport analysis demonstrates that it will not transport to the screens or core.

6.2.1.8.2.3 Coatings

To determine if the AP1000 design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to protective coatings (paint) in containment, the staff reviewed the information in the DCD and supporting documents according to the guidance listed in Section 6.2.1.8 of this report. The following are key guidance documents for coatings debris, and the first two are exclusive to coatings debris:

- "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated April 6, 2010 (ADAMS Accession Number ML100960495)

- Enclosure 2, “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation,” to “Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, ‘Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,’” dated March 28, 2008 (ADAMS Accession Number ML080230234)

- NEI 04-07, Revision 0 dated December 2004, and the staff’s accompanying safety evaluation

Sections 6.1.2 and 6.3.2 of DCD Revision 17 describe the selection and use of coatings for the AP1000. The three types of paint coatings in containment are epoxy, inorganic zinc, and unspecified manufacturer standard coatings on engineered components. The coatings are applied to three main types of surfaces inside containment: the inside surface of the containment shell, engineered components, and a variety of other surfaces. These other surfaces include both steel structures (walls, ceilings, floors, columns, beams, braces, and plates) and concrete structures (walls, ceilings, and floors). The plant design intentionally limits the amount of painted surface area within the ZOI of a LOCA jet.

The AP1000 design is similar to operating reactors with respect to coatings debris generation and transport. However, design features of the AP1000, such as the lack of containment spray during a LOCA and the low flow rates in the water pool are expected to reduce coatings debris generation and transport relative to operating reactors. For example, stainless steel is used to reduce the need for coatings. According to the descriptions in Section 6.3.2.2.7.3 of the DCD, stainless steel is used rather than coated carbon steel on surfaces within the coatings ZOI near the recirculation screens. The quantity of coatings is also reduced with respect to the originally certified AP1000 design by eliminating inorganic zinc as a primer for epoxy. Instead, inorganic zinc is used in the containment only on the inside surface of the containment shell and where the temperature during normal operating conditions exceeds the limit for epoxy.

The inorganic zinc coating inside containment is classified as safety Service Level I as defined in RG 1.54, Revision 1, which has guidance for procurement, application, inspection, and monitoring based on NRC-approved ASTM standards. The quality assurance program for these coatings meets the relevant requirements of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50. Inside containment, self-priming, high-solids epoxy is used on steel structures (walls, ceilings, floors, columns, beams, braces, and plates) and concrete structures (walls, ceilings, and floors). Appendix B to 10 CFR Part 50 applies to the procurement of these coatings. As described in DCD Sections 6.1.2.1.6 and 6.1.3.2, COL applicants are responsible for the programs to address quality assurance. Section 6.1.3.2 requires COL applicants to provide a program to control the procurement, application, inspection, and monitoring of Service Level I and III coatings, as well as most Service Level II coatings inside containment. The following paragraphs discuss these requirements in more detail below.

The ZOI for coatings conforms to the most recent (April 6, 2010) staff guidance on coatings evaluations for resolution of GSI-191. Specifically, the ZOIs are four times the pipe diameter (4D) for epoxy coatings and 10 times the pipe diameter (10D) for inorganic zinc coatings (DCD Section 6.3.2.2.7.1). The applicant determined the quantity of epoxy and zinc coatings by estimating the surface area with the limiting HL and CL ZOIs, and then applying the thickness and density in the plant specifications already developed. The methodology for selecting pipe break locations to calculate debris quantities is described in Section 6.2.1.8.2.1.1 of this report.

Briefly, the limiting CL location was the terminal end at RCP #2, in the west loop compartment. The limiting HL break was the terminal end at either steam generator, due to similarity between the compartments.

Using plant layout information, the applicant then identified the location of coated surfaces, such as walls, plates, pipes and beams with respect to each postulated ZOI. As explained in its June 30, 2010, Revision 2, response to RAI-SRP6.2.2-SPCV-25, the applicant calculated epoxy coating debris quantities of [ ] and [ ], respectively, for the limiting CL and HL breaks. For these locations, no inorganic zinc coating was located within the 10D ZOI. The staff performed confirmatory calculations based on the applicant's surface area, density, and thickness values. The staff concluded the assumed thickness value (2 coats at 0.01524 cm (0.006 in) each) was conservative for calculating coating weight because it matched the high end of the recommended thickness for nuclear-grade epoxy coatings. The staff also concluded the density value ([ ]) was conservative for calculating coating weight because it was higher than the minimum value (1602 kg/m^3 (100 lb/ft^3)) required by the design.

The applicant's analysis of coating debris quantity is based, in part, on a design change from coated carbon steel to uncoated stainless steel for ASME Code Section III steam generator system instrument piping. This change was described in the June 30, 2010, response, to RAI-SRP6.2.2-SPCV-25 and was submitted to the NRC in a letter dated July 8, 2010. The staff's evaluation of this proposed piping change is described in Section 23.P of this report. The assumed form of the coating debris also conforms to the staff guidance on coatings discussed above. Outside the ZOI, epoxy coatings are assumed to fail as chips and not transport to the screens, as discussed below. Inorganic zinc outside the ZOI is assumed to remain intact because of its Service Level I designation, which conforms to the staff guidance for qualified coatings outside the ZOI. Coatings within the ZOI are assumed to fail in the form of fine particles that transport both to the screens and core. The staff believes it is necessary to treat the coatings in the ZOI as fine particles for two reasons. First, coating chips are assumed not to transport, as discussed below. Therefore, only fine particles of coating would contribute to the debris loading at the screens and fuel assemblies. Second, as explained in the staff's March 28, 2008, guidance for coatings debris, it is important to treat coating debris from the ZOI as particles if there is a possibility of a thin, filtering bed. Since fuel assembly testing indicated head loss resulting from a filtering bed, it was appropriate to consider coating debris as fine particles that transport.

The assumption that Service Level II epoxy outside the ZOI will fail in the form of chips conforms to the guidance because the coatings will be treated as degraded qualified coatings. These coatings are procured (with one exception, as noted in DCD Section 6.1.2.1.6) under Appendix B to 10 CFR Part 50 and are tested under DBA conditions. However, the Service Level II coatings will not have the same level of quality assurance and assessment requirements as fully qualified coatings in containment (i.e., Service Level I). The staff determined that it is appropriate to treat this combination of product qualification and subsequent quality assurance as a degraded qualified coating for debris analysis and assume these coatings in accordance with the staff's March 2008 guidance. The exception to the procurement qualification is for the epoxy coatings in the CVS room that connects to the containment only through a drain line that discharges to the waste processing system below and away from the recirculation screens. As explained in DCD Section 6.1.2.1.6 and Table 6.1-2, the epoxy coatings in this room are not required to be procured under Appendix B to 10 CFR Part 50.

Debris in the form of chips is assumed not to transport to the screens because of the coating's high density, which is a design requirement. Design features of the AP1000 are also expected to keep debris from entering the pool near the screens. Coatings are required to have a density of at least 1.6 grams per cubic centimeter (g/cm^3) (100 lb/ft^3) (Epoxy will be purchased as DBA qualified. Epoxy outside the ZOI is treated according to the Keeler & Long tests for operating reactors and assumed to fail as chips. This testing (Keeler & Long Report No. 06-0413) was performed in support of GSI-191 resolution for operating plants and simulated debris generation from DBA-qualified epoxy and inorganic zinc coatings. The material tested was in the form of epoxy chips with an attached inorganic zinc primer. Nearly all of the epoxy remained as chips larger than 0.79 millimeter (mm) (1/32 in) in diameter, while the inorganic zinc failed as particulate and disbonded from the epoxy. The NRC guidance noted above recommends using these data in conjunction with coating chip transport data to reduce the amount of degraded qualified coatings assumed to transport to the screens.

The staff guidance dated March 28, 2008, states that if less than 100 percent transport of the coatings debris is considered, the basis for the debris settlement should be provided, such as the NRC-sponsored coating chip transport testing described in NUREG/CR-6916, "Hydraulic Transport of Coating Debris," issued December 2006. This testing found that at a steady-state velocity of 0.06096 meters per second (m/s) (0.2 feet per second (ft/s)), most epoxy coating chips with a density (2002 kg/m^3 (125 lb/ft^3)) similar to the specified density for AP1000 coatings ([ ]) did not transport to the end of the test flume at a water velocity of 0.06096 m/s (0.2 ft/s). The calculated maximum approach velocity range for the AP1000 is less than [ ] even for the most limiting case, which occurs [ ]. Therefore, the test conditions in NUREG/CR-6916 bound the AP1000 flows.

With respect to density, Service Level II coatings on structures in the AP1000 containment (except in the CVS room described above) and on engineered components in defined areas are required to have a density of 1602 kg/m^3 (100 lb/ft^3) in order to take credit for settling. The defined areas for engineered components are locations below the maximum flood level of a design-basis LOCA or above the maximum flood level and not inside a cabinet or enclosure. This requirement appears in DCD Tier 1, Table 2.2.3-4. As explained above and in DCD Section 6.3.2.2.7.1, containment recirculation screens have a debris curb as well as protective plates that extend at least 3.04 m (10 ft) in front and 2.13 m (7 ft) to the side of the recirculation screens. Considering these design features, the low fluid approach velocities at the screens, the specified coating density, and the NUREG/CR-6916 test data, the staff finds it reasonable to assume for the AP1000 plant design that coating chips (generated outside the ZOI) will not transport to the containment recirculation screens.

Similarly, for the IRWST, since wetted surfaces inside the IRWST are made of corrosion-resistant materials (e.g., stainless steel); no paint coatings are used on surfaces near the IRWST screens. As described in DCD Section 6.3.2.2.7.2, the IRWST is covered during operation, and the bottom of each vertically oriented screen is 15.24 cm (6 in) above the floor. Therefore, the only route to the IRWST for coatings debris is through the gutter system. However, because of the location of the gutter, the gutter trash rack, and the gutter discharge piping and the high density of the coatings, it is reasonable to assume that coating debris chips will not transport to the IRWST screens. Therefore, based on chip density, transport distance, and water velocity, the staff finds it acceptable to assume for the AP1000 plant that the epoxy chips will not transport to the containment recirculation or IRWST screens.

The staff's evaluation included review of new head loss testing for the recirculation screens, IRWST screens, and fuel assemblies. The testing performed for the screens and fuel assemblies included the amount of ZOI coatings. These tests used silicon carbide particles, with an average size of [ ], as the surrogate for coatings debris. As stated in the safety evaluation for NEI 04-07, the staff found it reasonable to treat coating debris as highly transportable particulates in the size range 10 to 50 microns where there is a possibility of a thin fiber bed. This particle size is based on the basic material constituents for inorganic zinc and epoxy coatings. Section 6.2.1.8.2.8 of this report discusses the test program and results.

For the reasons discussed above, the applicant's assessment using the NRC guidance in RG 1.82, Revision 3, for protective coatings is acceptable with respect to the ZOI, quantity and form of coating debris assumed, and the representation of coatings debris in the screen and fuel assembly testing. On this basis, the staff concludes that the AP1000 plant design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5), as they relate to the effect of protective coatings debris on long-term cooling following a LOCA, including specific consideration of the effects of protective coating debris under accident conditions.

6.2.1.8.2.4 Chemical Effects

To determine the compliance of the AP1000 design with the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to chemical debris formed in the post-LOCA containment pool, the staff reviewed the information in the DCD and supporting documents using the guidance listed in Section 6.2.1.8 of this report. The following are the key guidance documents for chemical debris, and the first two are exclusive to chemical debris:

- Enclosure 3, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated March 28, 2008 (ADAMS Accession Number ML080230234)

- "Final Safety Evaluation by the Office of Nuclear Reactor Regulation: TR WCAP-16530-NP, 'Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids To Support GSI-191,'" dated December 21, 2007 (ADAMS Accession Number ML073521294)

- NEI 04-07, Revision 0 and the staff's accompanying safety evaluation.

The staff's review of the applicant's supporting documents included an audit of detailed analyses and a calculation note documenting that applicant determined the chemical debris constituents and quantities using APP-PXS-M3C-052, "AP1000 GSI-191 Chemistry Effects Evaluation," Revision 1, dated May 4, 2009.

The applicant calculated debris from chemical precipitation using the methodology in WCAP-16530-NP. This methodology was developed for operating pressurized-water reactors (PWRs), and the safety evaluation listed above documents the staff's evaluation of the methodology for operating reactors. The staff's review of the methodology finds that the methods of WCAP-16530-NP also apply to the AP1000 based on the containment materials, the sump pH transient, and the pH buffering agent (trisodium phosphate (TSP)). With respect to temperature, however, the applicant noted in Section 2.4 of TR-26 that the predicted sump pool temperature for the AP1000 is briefly outside the range evaluated for WCAP-16530-NP (60 °C to 132.2 °C (140 °F to 270 °F)) but concluded that the methodology was still applicable. The staff reviewed the amount and duration of the temperature deviation during the audit of the calculation note referenced above. The staff estimated that the temperature was between [ ]. The staff concluded that it was appropriate to apply the release rate equations in WCAP-16530 during this period because of the short time relative to the 30-day event and because corrosion data for [ ] did not indicate a sharp increase in the corrosion rate in this temperature range (see Reference 6.2-3 in WCAP-16530-NP).

The amount of chemical precipitation predicted by the applicant was small relative to typical operating plants because of the AP1000's design features. The lack of fiberglass insulation eliminates a potential source of dissolved silica that could precipitate as silicate compounds. Because the AP1000 contains no calcium silicate insulation, there is no precipitation of calcium phosphate resulting from interaction of calcium silicate and the TSP pH buffer. For operating reactors with TSP buffer, calcium phosphate is one of the key chemical effects. The AP1000 analysis assumed calcium phosphate would be generated based on interaction between the dissolved TSP and calcium dissolved from the concrete. The quantity of $Ca_3(PO_4)_2$ precipitate was determined by first calculating the amount of concrete dissolution using the methodology in WCAP-16530-NP and then assuming all of the dissolved calcium precipitated as $Ca_3(PO_4)_2$. In its safety evaluation of WCAP-16530-NP, the staff concluded that this approach is conservative.

Testing and analysis for operating PWRs have shown that aluminum corrosion is a key contributor to chemical debris in the presence of an alkaline water pool. For the AP1000, the only intended use of aluminum is the ex-core detector housings, and the AP1000 design specifies stainless steel or titanium covers to isolate the aluminum from the water. The design includes an ITAAC to verify that the detectors are enclosed in stainless steel or titanium (DCD Tier 1, Table 2.2.3-4, Item 8.c). Although these covers eliminate the potential chemical interaction between aluminum and alkaline water during accident conditions, the applicant anticipated that some quantity of aluminum interacting with the water may be unavoidable in certain engineered components. Therefore, the applicant set an upper limit for this aluminum (27 kg (60 lbs)) in the design (DCD Section 6.1.1.4) and evaluated this quantity for aluminum corrosion and precipitation using the WCAP-16530-NP methodology. In its May 13, 2009, response to RAI-SRP6.2.2-CIB1-21, the applicant stated that the actual quantity of this aluminum would be tracked in accordance with a design calculation note. The staff finds the approach acceptable because the certification documentation in Tier 1 and Tier 2 of the AP1000 DCD limits the amount of wetted aluminum to the amount analyzed for chemical debris.

The applicant's analysis produced the following calculated mass of chemical debris:

Aluminum oxyhydroxide (AlOOH) [ ]
Sodium aluminum silicate ($NaAlSi_3O_8$) [ ]
Calcium phosphate ($Ca(PO_4)_2$) [ ]
TOTAL [ ]

In the WCAP methodology, these precipitates formed from aluminum released from metallic aluminum and concrete, silicon released from silica powder (Min-K insulation) and concrete, and calcium released from concrete. With respect to the total amount of precipitate, WCAP-16530-NP assumes all dissolved calcium, in the presence of phosphate, and all dissolved aluminum form precipitates. In its safety evaluation on WCAP-16530-NP, the staff found that this is a reasonable assumption for calcium and a conservative assumption for aluminum.

The staff reviewed the basis for the release of chemicals and precipitation of debris during its May 7, 2010, audit of APP-PXS-M3C-052, Revision 1. For example, the applicant assumed the coating was removed from all concrete in the flooded zone, and the WCAP release rate equation included the corresponding concrete surface area. For insulation in which silicon is released from silica powder during accident conditions (i.e., Min-K insulation), the release equation assumed that [ ] of the insulation was washed into the pool and subject to chemical release. The staff finds this is a reasonable assumption, since non-reflective metallic insulation (RMI) insulation in containment must be enclosed in seal-welded stainless steel and qualified through NRC-approved testing to show that a LOCA would not generate chemical debris (see DCD Section 6.3.2.2.7.1).

For aluminum, the applicant assumed a surface area of [ ]. The staff considered this a reasonable assumption because it corresponds to a relatively thin sheet of aluminum, on the order of 0.25 cm (0.1 in). This results in a relatively high calculated aluminum release because the release rate is proportional to surface area, and lower thickness corresponds to higher surface area for a given mass. In addition, it is the same conversion factor used in the spreadsheet in Appendix D to WCAP-16530 for calculating the mass of aluminum released. The staff used these assumptions, along with the temperature and pH profile provided in the calculation note, to perform independent confirmatory calculations using the WCAP-16530 spreadsheet. The staff also performed the calculations manually using the release rate equations derived in WCAP-16530-NP and incorporated into the spreadsheet. The staff's calculations produced the same values as the applicant's.

The AP1000 analysis assumes that all chemical debris transports (see Table 3-4 in TR-26). Therefore, the screen and fuel assembly testing included the entire chemical debris load, represented by AlOOH surrogate. The AlOOH precipitate is [ ] of the calculated mass of chemical debris for the AP1000. According to the test reports for screen and fuel assembly testing (WCAP-16914 and WCAP-17028, respectively), the surrogate chemical debris was prepared and added according the WCAP-16530-NP procedures. The staff's safety evaluation on WCAP-16530-NP states that surrogate precipitate prepared in accordance with the directions on WCAP-16530-NP provides adequate settlement and filterability characteristics to represent post-LOCA chemical precipitates in strainer head loss tests. The AlOOH precipitate was generated outside the test loops for the design-basis tests. In some supplemental (engineering) tests, the raw chemical components were added to the loop, and the precipitates formed in situ. The debris was added both sequentially, according to the WCAP, and coincidentally, to address whichever case is more limiting for the AP1000. The quantity of debris is intended to represent the generation and transport rate in the plant and meets the staff's March 28, 2008, guidance on chemical effects.

For the reasons discussed above, the staff concluded that the applicant's assessment of chemical effects conforms to the staff's March 2008 guidance with respect to the type and quantity of chemical debris, as well as the representation of chemical debris in the screen and fuel assembly testing. Therefore, the staff concludes that the AP1000 plant design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5), as they relate to the effect of chemical debris on long-term cooling following a LOCA, including specific consideration of the effects of chemical interaction and formation of debris under accident conditions.

6.2.1.8.2.5 Latent Debris

Latent or resident debris is dirt, dust, lint, and other miscellaneous material that is present inside containment during operation. As stated in DCD Tier 2, Section 6.3.2.2.7.1, Item 12, the design basis for the total amount of resident debris inside the AP1000 containment is 59.0 kg (130 pounds-mass (lbm)), of which up to 3.0 kg (6.6 lbm) is fiber. Additionally, COL Information Item 6.3-1, discussed in DCD Tier 2, Section 6.3.8.1, requires that COL applicants referencing the AP1000 design commit to a cleanliness program that limits the latent debris inside containment to these same quantities.

In TR-26, the applicant explained that limiting the total latent debris inside containment to 59.0 kg (130 lbm) is consistent with the practice for operating reactors. The information regarding latent debris in operating reactors came from publically available responses by individual plants to generic letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” dated September 13, 2004. The plants generally identified two quantities for latent debris: a walkdown value representing an estimate of the debris inside the plant based on physical sampling of containment surfaces, and an analysis (or bounding) value used in the GL 2004-02 evaluation. In accordance with Table 2-1 of TR-26, the average amount of latent debris in the 34 cited plants was 40.8 kg (90 lbm) based on walkdown data and 73.5 kg (162 lbm) based on the bounding values. The applicant attempted to correlate the amount of latent debris to the dominant type of insulation used in the plant (RMI or fibrous) and to the plant size, but it determined that neither was a strong predictor for the quantity of latent debris. The applicant concluded that other factors, such as the utilities cleanliness program, would be more indicative of the amount of debris found in the plant but did not provide additional data.

The staff’s review of TR-26, Table 2-1, found that while the values were consistent with the individual plant responses to GL 2004-02, the table only included data from half of the 69 operating PWRs. Therefore, while the reported averages are not representative of the entire PWR fleet, the data demonstrate that some operating PWRs do have latent debris bounding values of less than 59.0 kg (130 lbm). The staff emphasizes the bounding values over the walkdown values because, while some plants performed rigorous walkdowns in accordance with the guidance in the NEI 04-07 safety evaluation, other plants recognized that their walkdown calculations were not bounding; therefore, margin was added to the results before performing the analysis.

Rather than defining the fiber quantity as a mass percentage of total latent debris, the applicant has set the design basis for fiber inside the AP1000 containment to 3.0 kg (6.6 lbm), which is 5.1 percent of the total latent debris. In TR-26, the applicant explained that this is consistent with data from Table 2 of NUREG/CR-6877, “Characterization and Head loss Testing of Latent Debris from Pressurized-Water-Reactor Containment Buildings,” issued July 2005, which shows that the percentage of fiber in two of the four sample plants is less than 4 percent. The NEI 04-07 safety evaluation referenced this same NUREG to support the recommendation to treat 15 percent of the total latent debris mass as fiber. The different conclusions result from two different treatments of the NUREG-reported fiber-to-particulate compositions. In the data referenced by the NEI 04-07 safety evaluation, objects larger than a 0.335-cm (0.132-in) mesh sieve were removed from the sample because they were assumed to be nontransportable. In the data referenced in TR-26, the assumed nontransportable objects were retained because the applicant assumed that all latent debris is transportable and did not anticipate that utilities will remove the larger pieces of debris during sampling. While these assertions seem reasonable, the staff finds the design-basis quantities of total latent debris and fibrous latent debris inside the AP1000 containment to be acceptable because of the COL commitment to limit latent debris to the design-basis amounts. In accordance with the containment cleanliness program, applicants referencing the AP1000 must include a program to limit the amount of latent debris left inside containment following refueling and maintenance outages to 59.0 kg (130 lbm) total latent debris, of which up to 3.0 kg (6.6 lbm) is composed of fibrous material.

As described in Section 3.5.2.2.2 of NEI 04-07 and its related safety evaluation, miscellaneous debris such as equipment tags, tape, and stickers or placards affixed by adhesives that could become transportable should be considered part of latent debris. In accordance with the NEI 04-07 safety evaluation, if the lanyards or adhesives fail and the signs are transported to the screen intact, the available screen surface area should be reduced by an appropriate amount, which is called the sacrificial screen area. No sacrificial screen area is used in the AP1000 evaluation because the DCD requires that all potential sources of transportable material (such as caulking, signs, and equipment tags) be designed so they do not produce debris that will be transported to the filtering areas.

Specifically, unless the items are located inside a cabinet or enclosure, either they must be high density or testing must be performed at conditions that bound the AP1000 to demonstrate that the debris do not transport to any AP1000 filtering location or generate chemical debris. If testing is performed, the NRC must approve it.

6.2.1.8.2.6 Debris Transport

Debris transport is the estimation of the fraction of debris that is transported to each of the filtering locations (screens or core), considering blowdown transport, washdown transport, pool fillup transport, and recirculation transport. In the AP1000 design, washdown transport does not come from containment sprays but from PCS operation, where the majority of the condensed steam runs down the containment walls and the remainder falls from the containment dome. Debris transport is discussed in Section 3 of TR-26 and is evaluated as follows considering Section 3.6 of NEI 04-07 and the related safety evaluation.

All coatings in the ZOI are assumed to fail as fine particles and transport to the screens and core. Coatings outside the ZOI are assumed to remain intact or fail as chips that do not transport because of the high density. Section 6.2.1.8.2.3 describes coatings debris in more detail.

All of the chemical debris is assumed to transport to the screens and core. Section 6.2.1.8.2.4 describes the chemical debris in more detail.

The AP1000 analysis conservatively assumes that 100 percent of the latent debris in the containment will be transported by the fluid streams to the recirculating pool and remain entrained until it reaches the containment recirculation screens. The analysis assumes that 50 percent of the fibrous portion of the latent debris will transport to the IRWST screens. The applicant stated that this is conservative because the only path for debris to enter the closed IRWST tank is through the gutters, located just below the operating deck elevation along the containment walls. Debris transport from the operating deck to the gutter is limited by a border plate around the edge of the operating deck and a physical gap between this border plate and the gutter. Any fluid that makes it past the border plate but not across the gap will drain to the sump. Additionally, the operating deck is flat and has other unobstructed openings where water can spill directly into the containment sump. Because much of the latent debris in the AP1000 containment is located below the operating deck, it is not physically possible for it to be washed into the IRWST gutters. The staff agrees that the assumption of 50 percent transport of fibrous latent debris to the IRWST is conservative based on the IRWST gutter and operating deck configuration. During RNS operation in the recirculation mode, it is possible that some fluid will backflow through the containment recirculation screen to the IRWST. It is expected that the containment recirculation screen will capture the fibrous portion of the debris as it backflows, but not necessarily the particulates. To conservatively account for this phenomenon, the applicant assumed that the IRWST screens see 100 percent of the particulate portion of latent debris.

In the AP1000 design, MRI debris is not transported to the screens or core because of the low velocities of the recirculating water. Based on testing described in NUREG/CR-6808, a velocity of at least 0.061 m/s or 3660 liters per minute per square meter (Lpm/m^2) (0.2 ft/s or 90 gpm/ft^2) is required to move a settled piece of crumpled MRI debris that is 1.3 cm by 1.3 cm (0.5 in by 0.5 in). The maximum fluid approach velocities at the containment recirculation and IRWST screens that occur during RNS operation and as shown in Table 5-2 of WCAP-15914-P are significantly less than this value. The flow through the containment recirculation corridor, which is identified as the limiting area approaching the containment recirculation screen in the Revision 1 response to RAI-SRP6.2.2-SPCV-13, is also below this value.

In order to limit debris introduced during pool fillup, the AP1000 design restricts fibrous material in the containment outside the ZOI but below the maximum floodup level. Insulation located here must be MRI, jacketed fiberglass, or a suitable equivalent. Also, other potential sources of fibrous material such as ventilation filters or fiber-producing fire barriers are not permitted below the floodup level. The MRI and jacketed fiberglass insulation will not be dislodged or eroded by the recirculation flow rates and, thus, will not generate debris. A suitable equivalent insulation is defined as one that has been tested at conditions that bound the AP1000 operation to demonstrate that recirculation flows will not generate chemical debris, and if any other type of debris is generated, it must not be transportable. The NRC must approve test applicability and any subsequent analysis.

Other potential sources of nonfibrous transportable material in containment, such as caulking, signs, and equipment tags, are required to be either made of high-density material, located in an enclosure, or demonstrated to not transport during testing. If they are made of high-density material, which is defined as greater than 1.6 g/cm^3 (100 pounds per cubic foot (lb/ft^3)), it is expected that they will settle and not be carried by the low AP1000 velocities. The minimum 2.5-hour delay between the accident and start of recirculation provides a reasonable amount of time for debris to settle on the containment floor.

While the applicant considered washdown from condensed steam running down the containment walls, it did not consider washdown associated with the containment spray system, which could transport more debris into the fluid stream. The staff agrees that it is not necessary to consider the impact of the containment spray system because it is a nonsafety-related system that will only be used in the event of a severe accident. During a severe accident, core heat removal or coolant has already been lost, and the containment spray's effect in transporting additional debris is not significant. Also, inadvertent actuation of the containment spray system during power operations has previously been found to be noncredible, as described on pages 6-25 of NUREG-1793.

For the AP1000, some break locations will be submerged once the containment has flooded up during the post-LOCA recirculation core cooling phase. When the containment water level is at the elevation of the break, in addition to the flow through the PXS intact DVI recirculation lines, flow carrying unfiltered debris can enter directly into the reactor vessel through the submerged break. A DECL break or DEHL break at the reactor vessel and a DEDVI line break can be submerged when the final containment floodup water level is reached. Any one of these breaks could allow direct transport of unfiltered debris into the reactor, bypassing the containment recirculation screens, during the long-term cooling phase. The DECL break and DEDVI break could allow unfiltered debris into the downcomer, lower plenum, and ultimately into the core, potentially resulting in blocking reactor coolant flow through the core, and the DEHL break allows unfiltered debris into the upper part of the core area.

Of the DEDVI and DECL breaks, the DECL break potentially provides the greatest amount of unfiltered debris to the core inlet and the highest percentage of debris bypass (i.e., the percentage of the debris that could be transported into the RCS without filtering by the IRWST and containment recirculation screens). The applicant determined the percent of debris bypass for the DEDVI break and DECL break by calculating the flow split between the recirculation flows through the break and through the PXS intact DVI lines. Section 3.3 of TR-26 discusses debris transport to the core and calculations to determine the flow split (i.e., the percentage of recirculation flow entering the reactor vessel through the break carrying unfiltered debris). For the flow split calculation, the applicant conservatively assumed that [ ] would transport the entire available debris load to the recirculation screens and into the reactor vessel through the break and ultimately to the core. Therefore, the flow split is determined by integrating the relative recirculation flows through the break and through the PXS for [ ] at the containment floodup water level elevation. Flow rates through the break and through the PXS were obtained from WCOBRA/TRAC long-term cooling cases. The percentage of the water mass that has entered through the flooded break and through the PXS is determined after [ ]. The applicant performed the calculation for a DEDVI LOCA and for a DECL LOCA at the reactor vessel nozzle. In RAI-SRP6.2.2-SRSB-41, the staff asked that the applicant clarify the flow split discussion for the DECL and DEDVI breaks and related calculation presented in TR-26. The applicant responded by revising TR-26 to provide more information concerning the DEDVI and DECL flow splits and break calculations in Tables 3-1 and 3-2, respectively.

For a DECL break, the flow split would be greater than 10 percent flow through the PXS recirculation flowpath and [ ] percent flow through the rupture in the CL, which is greater than [ ] percent for the DEDVI break. The applicant assumed that: (1) 90 percent of the containment water would flow through the break in the CL; and (2) 90 percent of the fiber debris and 100 percent of the particulate debris in the containment would directly enter the reactor core through this DECL break. The applicant used the 90 percent unfiltered fibrous debris assumption in its fuel assembly testing, documented in WCAP-17028-P, by estimating a total fiber load in the containment of 2.99 kg (6.6 lbs) and multiplying that amount by 90 percent, yielding an estimated 2.69 kg (5.94 lb) of fiber directly entering the reactor core.

The staff also evaluated the debris transport through a HL break. In RAI-SRP6.2.2-SRSB-31, the staff asked the applicant to clarify the reasons why the DEHL break is not the most limiting break with respect to debris transport. Through testing and thermal hydraulic evaluations, The applicant developed the probable scenario that occurs during a DEHL break. During a postulated HL break, containment floodup water carrying unfiltered debris begins to enter the top of the core region through the submerged break, and water filtered by the IRWST and recirculation screens also flows into the downcomer and into the reactor core inlet through the two intact DVI lines. The debris that enters the upper core region through the break could block the upper core area, and some could be discharged through the automatic depressurization system (ADS) Stage 4 (ADS-4) valves connected to the HLs. The debris that is not discharged from the RCS through the ADS-4 valves can enter the upper portion of the core and flow down though peripheral low-power fuel assemblies. In its evaluation, the applicant identified that at some point, because of the cross-flow thermal hydraulics in the AP1000, the downflow of water through the peripheral fuel assemblies will cross into the hotter fluid that is rising through the core, and the debris carried by the water that entered the break will mix with the other filtered water flowing into the core. As a result, some of the debris from the HL break could be captured on the upper tie plates of the core.

In its response to RAI-SRP6.2.2-SRSB-31, the applicant further reported that it conducted fuel assembly debris-loading head loss testing to estimate the amount and impact of debris that could be captured in the upper core region. WCAP-17028-P, designating the tests evaluating HL flow conditions as tests 35, 38, and 39. The applicant demonstrated through the fuel assembly testing that on the top of the fuel assembly debris beds will not form in the presence of two-phase flow, which would be prevalent in the AP1000 core during long-term cooling. As a result of this test, the applicant concluded that fibrous debris that enters through a HL break would ultimately be purged and captured by the IRWST and containment recirculation screens, thus purging the upper core region of debris over time during long-term cooling. The applicant further stated that debris that breaks loose from the top of a fuel assembly would likely discharge through ADS-4 valves and subsequently be filtered by the IRWST and containment recirculation screens. Section 6.2.1.8.2.7 of this report presents more discussion of these breaks.

On the basis of its review of the RAI-SRP6.2.2-SRSB-31 response and the associated fuel assembly testing in WCAP-17028-P, the staff finds that debris plugging in the core from a DEHL break is not the most limiting break with respect to debris transport and accumulation.

In summary, in Table 3-4 of TR-26, Revision 8, the applicant presented the AP1000 licensing basis fibrous, particulate, and chemical debris that could be transported into the reactor vessel and potentially reach the fuel assemblies. These debris loads were based on a total latent debris load of 58.96 kg (130 lbs), a total ZOI coating fine particles of 31.75 kg (70 lbs), and a chemical debris load of 25.85 kg (57 lbs). Of the 58.96 kg (130 lbs) of latent debris, no more than 2.99 kg (6.6 lbs) was estimated to be fibrous.

In TR-26, the applicant reported that chemical debris load is based on the type and quantity of chemical precipitates that may form in the post-LOCA recirculation fluid for the AP1000 design. The evaluation presented above identified the DECL break as the worst case break that provides debris to the core.

Based on the DECL break flow split calculation performed by the applicant, 90 percent of the water that enters the reactor vessel during long-term cooling comes through the CL break and 10 percent comes into the reactor as filtered water through the intact DVI lines. For the purpose of these evaluations, the applicant assumed that 90 percent of all fiber and 100 percent of all particulates and chemicals in the containment are transported to the reactor unfiltered.

On the basis of its review of debris transport analysis to the core, the staff finds that the applicant has conservatively evaluated the quantity of debris transported and the minimum transport time to the reactor core.

6.2.1.8.2.7 Availability of Long-Term Core Cooling

The AP1000 PXS and containment system are designed to continuously provide adequate cooling of the reactor following a LOCA. After the initial injection of water from the core makeup tanks (CMTs) and the accumulators, the IRWST provides safety injection by gravity drain. As the IRWST injection continues, the containment sump water floods up above the reactor vessel. After the IRWST level drops to a low setpoint, the containment recirculation valves are open to provide the RCS with recirculation water from the containment sump to maintain core cooling indefinitely. During the long-term cooling phase of the LOCA, steam released from the reactor through the ADS-4 valves is condensed on the inner surface of the steel containment vessel, which is cooled on the outside by the PCS. The condensed water in the containment is collected in a gutter and returned to the IRWST. Both the IRWST and containment recirculation have screens to filter debris and protect the PXS flowpaths into the RCS.

Subsequent to a LOCA that occurs with the break below the post-LOCA water level in the containment, significant unfiltered debris can enter the reactor vessel through the submerged break location.

DCD Tier 2, Section 15.6.5.4C, describes the long-term cooling analysis to demonstrate that the passive safety systems provide adequate emergency core cooling system (ECCS) performance during the IRWST injection and containment recirculation duration. The analysis was performed for a DEDVI line break in the PXS room using the WCOBRA/TRAC code during the IRWST injection phase continuing into containment sump recirculation. When a quasi-steady state was achieved, a WCOBRA/TRAC window mode of calculation was performed. The boundary conditions for the WCOBRA/TRAC analysis, such as the containment pressure and the level and temperatures of the liquid in the containment sump, are based on the WGOTHIC calculations. The long-term cooling analysis demonstrated that: (1) the core remains cooled for the duration of the long-term cooling phase; (2) the boron concentration in the core keeps the core noncritical; and (3) there is no boron precipitation in the core during long-term cooling following a LOCA. Section 15.2.7 of NUREG-1793 describes the staff's evaluation of this long-term cooling analysis. However, the DCD long-term cooling analysis did not consider the effect of debris in the containment on long-term cooling or the GSI-191 evaluation of the AP1000.

As part of the GSI-191 review of the AP1000, the applicant evaluated the effects of containment debris on the long-term core cooling capability. In the AP1000 design, debris could block core flow and adversely affect long-term cooling following a LOCA in three locations: (1) the core inlet; (2) containment recirculation screens; and (3) IRWST screens. Core inlet head loss is the most significant parameter in these analyses because most of the long-term cooling flow and debris goes through the DEDVI and DECL breaks unfiltered into the downcomer and through the core inlet region, and because the AP1000 IRWST and containment recirculation screens are sufficiently large that the debris-induced head loss on the screens is very small. A DEHL break at the reactor vessel adds unfiltered debris into the top of the core region, which does not impact the core inlet loss coefficient. However, it has the potential of flow blockage at the upper part of the core.

The staff's evaluation includes: (1) long-term core cooling analysis sensitivity studies of the effects of the debris-induced head losses at the debris collection locations (i.e., the IRWST and containment recirculation screens) and the reactor core on the long-term cooling analysis and (2) head loss testing of the IRWST and containment recirculation screens and fuel assembly. The applicant's approach is first to perform the long-term cooling sensitivity analysis using the WCOBRA/TRAC code and using arbitrary, nonmechanistic head loss coefficients at the core inlet and the IRWST and containment recirculation screens to determine the maximum allowable head losses in the debris collection locations while still maintaining adequate long-term core cooling. The maximum allowable head losses thus determined serve as the acceptance criteria for the debris-induced head loss tests. The applicant then conducted the debris-induced head loss testing of the IRWST and containment recirculation screens and fuel assembly to demonstrate that, with the amount of debris collected in the IRWST and containment recirculation screens and the core, the respective test acceptance criteria would not be exceeded. Therefore, one can conclude that adequate core cooling is maintained in the presence of the containment debris. The staff's evaluation of the applicant's submittal appears below.

6.2.1.8.2.7.1 LTC Sensitivity Studies to Determine Adequate Core Cooling

The applicant performed long-term cooling sensitivity studies as documented in TR APP-PXS-GLR-001, Revision 4. As discussed in Section 6.2.1.8.2.1 of this report, Westinghouse selected the DEDVI break for Sensitivity Cases 1 through 10 for the long-term cooling evaluation following a LOCA. All cases were performed using the decay heat assumption in Appendix K to 10 CFR Part 50. As the same long-term cooling case analyzed in DCD Section 15.6.5.4C, Case 1 considers a DEDVI line break in the PXS room with a moderate increase in the core and screen pressure drops resulting from debris blockage. The flow resistance of the lower support plate at the core inlet is increased to model a dP of 3 ft of water at the DCD analysis flow rate of 68.9 kilograms per second (kg/s) (152 pounds per second (lb/s)). Cases 2 and 3 increase these pressure drops. Each of these cases includes added pressure drop across the core, recirculation screens, and IRWST screens. Cases 1, 2, and 3 were performed at the start of containment recirculation at 2.6 hours after a LOCA. By this time, only a small portion of the total debris would have been transported into the RCS, and the debris dP would not have increased to its maximum.

In the sensitivity studies and the containment recirculation sump and IRWST screen testing, the applicant also demonstrated that the screen head losses, considering debris amounts and types, flow rates, and screen size and type, were small, or less than a fraction of an inch of water.

For Sensitivity Cases 4 through 10, the core inlet loss coefficient was further increased to represent different levels of debris plugging at the core inlet. Case 10 had the highest core entrance loss coefficient. The IRWST and containment recirculation screen loss coefficients were assumed to be 0 because they are insignificant compared to the core inlet loss coefficient. Sensitivity Cases 4 through 10 were performed at a later time of 8.6 hours after a LOCA, which is considered the earliest time at which all the debris in the water of the containment can be transported from containment into the core, and increased the dP to its maximum.

In RAI-SRP6.2.2-SRSB-36, the staff asked the applicant to clarify its reason for running Sensitivity Cases 4 through 10 at 8.6 hours after a LOCA. In its response dated January 29, 2010, the applicant stated that for Cases 1 through 3, it assumed that peak core dP would occur earlier, before all particles and fibers would be transported to the core. Sensitivity Cases 1 through 3 were analyzed at 2.6 hours after a LOCA. At this time, only a small amount of debris would start to enter the core through the break. For Cases 4 through 10, the applicant assumed, for purposes of core inlet debris accumulation, that the [ ] would transport all of the particles and fibers. As subsequent fuel assembly head loss tests were run, the applicant noted that the maximum core dP occurred later than the evaluation time of 8.6 hours of debris transport. It therefore concluded that using the decay heat level at 8.6 hours after a LOCA in Sensitivity Cases 4 through 10 was conservative for evaluating fuel assembly test results where maximum core dP is reached 9 hours or later post-LOCA. In its response to RAI-SRP6.2.2-SRSB-40, the applicant demonstrated that subsequent fuel assembly concurrent debris addition test results yielded the peak core dP at much later times after the LOCA than

8.6 hours. Table 3-3 of TR-26, Revision 8, showed that for each of the later fuel assembly debris loading tests, peak core dP at an equivalent plant time far exceeded 8.6 hours after LOCA. The equivalent plant time was determined by comparing the amount of chemical addition before the occurrence of the peak dP in the test with the post-accident chemical effects evaluation. The staff agrees that the sensitivity cases calculated at 8.6 hours are conservative for assuming the maximum core inlet blockage because 8.6 hours is far earlier that the time of the peak debris-induced core inlet dP in the plant. As discussed in the Test Acceptance Criteria section, the core inlet dP for Case 10, which was a DEDVI line break in the loop compartment with the largest core inlet resistance among all sensitivity cases, will be used as the acceptance criterion for the fuel assembly debris blockage head loss tests after 9 hours. The core inlet dP for Case 3 will be used as an additional acceptance criterion for the fuel assembly head loss test before 9 hours.

For Sensitivity Case 11, the applicant modeled a DEDVI break in the loop compartment close to the reactor vessel DVI nozzle, simulating significant resistances at the core exit region. The added debris resistance was applied to the core exit in order to provide insights into the impact of containment debris entering the upper part of the core during a postulated HL break LOCA. Because postulated debris would be introduced into the upper plenum during a HL break scenario, no increase in the core entrance flow resistance above the value associated with normal plant power operation was modeled. This case was executed at the decay heat level at 8.6 hours post-LOCA time to transport sump debris into the reactor vessel. The results of Case 11 provide acceptable values of core flow and debris-induced dP for application to fuel assembly head loss testing that evaluated DEHL LOCAs.

The applicant used WCOBRA/TRAC and WGOTHIC for the RCS transient and containment analyses, respectively, of the long-term cooling analysis, as documented in DCD Tier 2, Section 15.6.5.4C, and for the 11 sensitivity cases evaluated in APP-GW-GLR-001. The long-term cooling analysis used a detailed nodalization model to represent the AP1000 core for the WCOBRA/TRAC analysis. WCAP-14776, "WCOBRA/TRAC, OSU Long-Term Cooling Final Validation Report," Revision 4, issued March 1998, documents the code verification for the long-term cooling analyses. The applicant used the WGOTHIC code, described in WCAP-15846, Volume 1, Revision 1, to calculate containment boundary conditions. The fan coolers were assumed to be operating to minimize containment pressure. The staff previously approved the application of WCOBRA/TRAC and WGOTHIC for long-term cooling calculations performed in the AP1000 DCD, as discussed in Chapter 21 of NUREG-1793.

The staff reviewed the application of WCOBRA/TRAC and WGOTHIC for performing the sensitivity studies documented in APP-GW-GLR-001. The staff conducted an audit in Monroeville, Pennsylvania, September 21–22, 2009, and a followup audit March 22–24, 2010. During the September 2009 audit, the staff identified discrepancies between the WCORBRA/TRAC and WGOTHIC calculations for sump water temperature input. In RAI-SRP6.2.2-SRSB-25, the staff asked the applicant to address these inconsistencies.

In its response, the applicant stated that the WGOTHIC containment analysis case was reanalyzed using corrected revised input, and the calculated sump water temperatures were within -17.22 °C (1 °F) of the corresponding values presented in the WCOBRA/TRAC calculation. The applicant stated that since the boundary conditions for WCOBRA/TRAC from the new WGOTHIC case differed minimally, the conclusions of Sensitivity Cases 1 through 3 in APP-PXS-GLR-001 were not affected.

During the audit performed March 22–24, 2010, the staff reviewed the new WGOTHIC analyses and results used for Sensitivity Cases 4 through 10. The applicant also responded to RAI-SRP6.2.2-SRSB-25, Revision 1, which addressed the WGOTHIC containment pressure and sump temperatures used in the Sensitivity Case 1 through 10 long-term cooling analyses. The pressure used in the long-term cooling calculation was 110 kPa (16 pounds per square inch absolute (psia)), which is lower than the pressure calculated by WGOTHIC. The sump temperatures calculated by WGOTHIC were slightly different from the sump temperatures used in the DCD Revision 17 long-term cooling calculations. The applicant has established that a lower pressure is conservative for long-term cooling calculations and that the small changes to the sump temperature do not impact the long-term cooling analysis. Based on its review of the new WGOTHIC analysis and the information provided in response to RAI-SRP6.2.2-SRSB-25, Revision 1, the staff concluded that the analyses were performed in a conservative manner and were acceptable.

The more limiting long-term cooling sensitivity cases in APP-GW-GLR-001 assume large core inlet head losses to simulate extreme core inlet blockage by the containment debris. These cases result in low core flow and high steam quality, which differ from the DCD long-term cooling analysis. The staff, in RAI-SRP6.2.2-SRSB-39, asked the applicant to confirm that the WCOBRA/TRAC code had been validated for the low flow and high steam quality conditions.

In its response dated January 29, 2010 (ADAMS Accession Number ML100330392), the applicant indicated that the validation of WCOBRA/TRAC is provided through the comparison of WCOBRA/TRAC simulations to test data from boiloff tests documented in WCAP-15644-P, “AP1000 Code Applicability Report,” Revision 2, issued March 2004. During the audit performed March 22–24, 2010, the staff reviewed the range of applicability of boiloff tests G1 (WCAP-9764, “Documentation of the Westinghouse Core Uncovery Tests and Small Break Evaluation Model Core Mixture Level Model,” issued July 1980) and G2 (Andreychek, T.S., “Heat Transfer above the Two-Phase Mixture Level under Core Uncovery Conditions in a 336 Rod Bundle,” Volumes 1 and 2, Electric Power Research Institute (EPRI) Report NP-1692, issued January 1981) to the conditions that would be simulated for Sensitivity Cases 4 through 10. In WCAP-15644-P, the applicant used WCOBRA/TRAC to simulate selected G1 and G2 boiloff tests, at low pressure, low flow and a range of power levels typical of AP1000 LTC conditions. WCOBRA/TRAC tended to overpredict the level swell in the core during boiloff scenarios. However, when a multiplier (YDRAD=0.8) is applied to the interfacial drag coefficient computed from the vertical flow regime models in the code, WCOBRA/TRAC generally predicted the level swell to within +/- 20 percent of the measured value. Therefore, the YDRAG=0.8 was selected for use in the WCOBRA/TRAC model for the AP1000 LTC analysis. The staff reviewed the ranges of the G1 and G2 test conditions and found that the conditions simulated for the G1 and G2 tests by WCOBRA/TRAC bounded the conditions calculated in Sensitivity Cases 4 through 10.

The staff noted in its review that the validation evaluations were performed using the WCORBRA/TRAC M7AR4_SB03 code version, whereas the current long-term cooling sensitivity analysis cases were performed using WCORBRA/TRAC M7AR7_AP. The staff asked if the difference in the different code versions could invalidate the conclusions of the previous studies to validate that WCORBRA/TRAC can successfully calculate conditions seen in the high debris blockage conditions of Sensitivity Cases 4 through 10. In Revision 2 of its response to RAI-SRP6.2.2-SRSB-39, dated June 28, 2010, the applicant indicated that it reviewed the differences between the two code versions and found that all but one of the error corrections and code updates can be classified into one of the four categories judged to have no or negligible impact on the applicability of the G1 and G2 validation calculations to the long-term cooling debris sensitivity cases. The only code difference judged to potentially affect the level swell calculations is related to the use of a level sharpener in the WCORBRA/TRAC M7AR4_SB03 code version. The WCOBRA/TRAC_AP code does not contain an explicit mixture level tracking model and, therefore, tracking of two-phase mixture level is accomplished by nodalization and prediction of the axial void gradient between hydraulic cells. The "level sharper" model in WCOBRA/TRAC M7AR4_SB3 code version was developed to locate the mixture level in the hydraulic cells where a sharp void fraction gradient is detected in the vicinity of the two-phase mixture level. The level sharpener logic is only applied to the void fraction used in the fuel rod heat transfer calculations in the hydraulic cells where detailed representation of local void fraction is important to assure that fuel rod heat transfer is computed based on the appropriate fluid condition. The level sharpener does not directly affect the global void fraction distribution. Since the level swell is a measure of two-phase mixture level relative to the collapsed liquid level, and is calculated in terms of the average void fraction, the precise mixture level has minimal effect on the level swell calculation. Therefore, the level sharpener model has negligible effect on the level swell calculation, and its impact on the G1 and G2 simulations is small. In other word, similar conclusions for the WCOBRA/TRAC code validation with the G1 and G2 simulations can be drawn with and without the level sharpener model. Therefore, the staff concludes that the use of the interfacial drag coefficient multiplier identified with the code version with the level sharpener model is applicable to the code version without the level sharpener logic. It should be noted that the WCOBRA/TRAC-M7AR4_AP code version used in the DCD Revision 15 long-term cooling analysis also did not include the level sharpener logic.

Sensitivity Cases 1 through 3 were analyzed with the AP1000 reference core design described in DCD Chapter 4. During the March 22–24, 2010, audit, the staff found that Sensitivity Cases 4 through 11, documented in Westinghouse calculation note APP-SSAR-GSC-732, "AP1000 AFCAP Post-LOCA Long-term Core Cooling Analysis," Revision 0, dated November 12, 2009, were performed with an advanced first core design, which differs from the DCD reference core design. Since the Advanced First Core Analysis Program (AFCAP) core design differs from the core design described in the AP1000 DCD, the staff questioned the applicability of the LTC analysis for cases 4 through 10 to the DCD core design. In response to RAI-SRP6.2.2-SRSB-42, the applicant describes the differences between the AFCAP core design and the DCD core design. The majority of the fuel assembly characteristics are either unchanged or have trivial changes in the AFCAP core design, [ ]. These changes are reflected in the input of the WCOBRA/TRAC AFCAP analysis for LTC sensitivity study Cases 4 through 10 using larger values of the flow resistance form loss coefficients for the mixing vane grids (MVG) and intermediate flow mixer (IFM) grids in the active fuel region. Other parameters in the WCOBRA/TRAC LTC analysis remain unchanged between the AFCAP and the DCD reference core design. The WCOBRA/TRAC LTC analysis was performed with the core geometry and form loss coefficients of fuel assemblies, plus a large non-mechanistic form loss coefficient at the core inlet to simulate possible worst case debris plugging. Since the core inlet flow resistance used in the LTC sensitivity runs are significantly larger than the flow resistances of the MVG and IFM, the differences in the MVG and IFM flow resistance between the AFCAP and DCD core design is not significant. In addition, the use of higher resistances for the MVG and IFM for the AFCAP core design would result in lower core flow rate than the DCD core design with lower MVG and IFM resistances. Since the objective of the LTC sensitivity studies is to establish the flow/pressure drop acceptance criteria for the AP1000 debris-induced core inlet blockage tests, the analysis using the AFCAP core design would result in lower core flow rate and inlet pressure drop. This results in more restrictive acceptance criteria for the fuel assembly debris-induced head loss tests. The use of the more restrictive acceptance criteria from the AFCAP core design for the fuel assembly head loss tests of the DCD referenced core design is conservative and is, therefore, acceptable.

LTC Sensitivity Study Results

Section 3 of APP-GLR-PXS-001, Revision 4, presents the results of the long-term cooling sensitivity analysis. For the DEDVI line break, the effects of increasing the flow resistance at the core entrance are generally reflected in the increase in the downcomer liquid level, a decrease in the core flow rate, a decrease in the core collapsed liquid level, an increase in the upper core void fraction, and an increase in the quality of the discharge flow through the ADS-4 valves. The upper plenum pressure for each case essentially reflects the containment pressure. In all cases analyzed, the collapsed liquid level in the core is higher than or near, the core midplane. The mean HL collapsed liquid level is above the HL centerline. For each sensitivity case, Table 4-1 in APP-GLR-PXS-001, Revision 4, provides the time after LOCA, core inlet loss coefficients, core flow rate, core inlet dP, ADS-4 steam quality, and maximum core boron concentration.

For Sensitivity Case 3, the upper plenum pressure of 22 psia is essentially unchanged from the DCD case. The downcomer liquid level has increased compared to the DCD case because of added core inlet flow resistance. Injection rates through the DVI lines into the vessel are reduced compared to the DCD case values. Flow through the intact DVI line is reduced to 25 kg/s (56 lbm/s) versus 35 kg/s (77.2 lbm/s) in the DCD analysis. Flow into the vessel through the broken DVI line is reduced from 34 kg/s (75 lbm/s) in the DCD analysis to 25 kg/s (55 lbm/s) in Sensitivity Case 3. The core flow is reduced to 50 kg/s (111 lbm/s) with a pressure loss of 24.1 kPa (3.5 pounds per square inch (psi)). The HL collapsed liquid level is above the HL centerline. The quality of the flow discharged through ADS-4 valves in Sensitivity Case 3 is approximately 0.35. The core boron concentration is 4,700 parts per million (ppm).

The thermal hydraulic behavior in the core was shown to be most limiting in Sensitivity Case 10. For Sensitivity Case 10, the downcomer liquid is established at a higher level than in the DCD analysis because of the greater added resistance at the core entrance. Boiling in the core produces steam and a two-phase mixture that flows out of the core into the upper plenum. The core collapsed liquid level is maintained at a mean level above or near the core midplane even with the added core entrance resistance. Boiling causes pressure variations, which in turn cause variations in the core collapsed level and the flow rates of liquid and vapor from the top of the core. The HL collapsed liquid level is around the HL centerline. The peak cladding temperature of the hot rod closely follows the saturation temperature. The flow through the core and out of the RCS provides adequate flushing to preclude the unacceptable concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the collapsed liquid level remains well above the active fuel length.

For Sensitivity Case 10, the upper plenum pressure reflects the prevailing containment pressure of 110 kPa (16 psia) calculated at 31,000 seconds. Injection rates through the DVI lines into the vessel are reduced compared to the DCD case values, and the injection flow rate is greater through the lower resistance broken DVI line than it is through the intact DVI line. The core flow is predicted to be 29.5 kg/s (65 lb /s), with a pressure loss of 28.3 kPa (4.1 psi). The steam quality of flow discharged through ADS-4 valves is approximately 0.49. This liquid carryover is adequate to limit the concentration of boric acid in the core water to a value of 6,100 ppm. This is lower than the maximum value of 7,400 ppm at the time of recirculation initiation in the DCD long-term cooling analysis that was shown to be acceptable in DCD Section 15.6.5.4C.4.

The applicant determined that Sensitivity Case 10, which is a DEDVI break in the loop compartment, represented the worst case condition assuming that all the debris will reach the core entrance sometime after one complete recirculation of the entire containment water volume.

The applicant used Cases 3 and 10 to establish the acceptance criteria for the AP1000 fuel assembly debris head loss testing. The staff's evaluation of these acceptance criteria is discussed in the Test Acceptance Criteria section of this SER section.

For Case 11, the resulting injection flow rate through the lower resistance broken DVI line 62.8 kg/s (138.5 lb /s) is greater than it is through the intact DVI line 34.5 kg/s (76 lbm/s). The core flow is predicted to be 97.3 kg/s (214.5 lb /s) with an average pressure loss of 13.8 kPa (2 psid) at the core exit where the added debris resistance was applied. The results of Case 11 also demonstrate long-term cooling performance comparable to the DCD long-term cooling case. The DEHL break acceptance criterion is core exit pressure drop of 13.8 kPa (2 psid) at a flow of 97.3 kg/s (214.5 lbm/s).

It should be noted that the WCOBRA/TRAC long-term cooling analysis did not consider the effects of the potential accumulation of noncondensable gases in high points in the PXS flowpaths. In a letter dated May 25, 2010, the applicant proposed the PXS design changes (Change No. 66), including installation of high point vents, to reduce the potential impact on gas intrusion in the PXS flowpaths. The staff considers that an appropriate PXS design change to prevent potential gas accumulation in the PXS system is an acceptable substitute for not considering the noncondensable gas effects in the long-term cooling analysis. The staff evaluation of DCP Change No. 66 will be addressed in Chapter 23.

6.2.1.8.2.7.2 Fuel Assembly Head loss Testing

The applicant performed a series of experiments to quantify the effect of fibrous and particulate debris and containment chemical effects on the head loss across the fuel assemblies of an AP1000 core during a postulated LOCA, documented in WCAP-17028, Revision 6, and performed in consideration of GSI-191. The objective of these experiments was to demonstrate that there is reasonable assurance that the AP1000 can provide adequate post-LOCA long-term core cooling. The applicant used a fuel assembly design that is consistent with the fuel assembly design described in the AP1000 DCD. The flow rates, debris loadings, and method of debris addition varied from test to test. The ratio of fibrous to particulate debris varied, as did the temperature and chemistry of the coolant. The purpose of the tests was to select a combination of debris variables and simulated plant variables that would bound any AP1000 LOCA and to demonstrate available long-term cooling margin in the AP1000 design such that with the flow blockage caused by the containment debris transported to the reactor vessel, the head losses determined by the fuel assembly testing are bounded by the test acceptance criteria described below.

The applicant performed 39 different experiments. Section 7 and Tables 7-1 through 7-3 of WCAP-17028 summarize the test matrix and initial conditions for the tests. Table 8-1 of WCAP-17028 summarizes the results of experiments that were performed with various debris loads, flow rates, fiber length, chemical effects, and fiber, particulate, and chemical addition sequencing.

The applicant had initially conducted the first 16 fuel assembly tests correlating to long-term cooling Sensitivity Cases 1 through 3. The applicant varied the particulate, fiber, and chemical amounts. The composition of the fibers for these 16 fuel assembly tests also varied. For the subsequent 23 fuel assembly tests, except for tests 35 and 38, the applicant increased fibrous debris loading based on the design-basis fiber amount of 3.0 kg (6.6 lb) in the containment, equivalent to 2.7 kg (6 lb) of fiber to the core inlet, as discussed in Section 6.2.1.8.2.6 of this report. The applicant chose debris loadings in the fuel assembly tests to bound the quantities that could be transported to one fuel assembly in the AP1000. Table 3-4 of TR-26 presents the AP1000 design-basis fibrous, particulate, and chemical debris that could be transported into the reactor vessel and possibly reach the fuel assemblies.

The applicant selected flow rates to bound the conditions expected during post-LOCA recirculation core cooling. Except for Tests 2 and 8 through 11, the first 16 tests were conducted with constant flow rates. Tests 17 through 39 were conducted with variable flow rates, with the flow reduction based on the increase in the dP resulting from debris additions.

Tests 1 through 21 and Test 23 were performed using sequential additions of particles, then fibers, then chemical surrogate. Test 22 and Tests 24 through 39 were performed with concurrent additions of particles, fibers, and chemical surrogate. The applicant prepared fibrous and particulate debris loads and chemical precipitates outside of the test loop and added them to the makeup tank in accordance with the applicant's test plan. All of the experiments included fibrous and particulate debris and chemical reaction products that were added to the makeup tank as water suspensions. The sequential and concurrent debris addition schemes are discussed in the Debris Addition Scheme section below.

The fuel assembly test loop consists of a scaled length of a fuel assembly inside a Plexiglas case, a mixing tank, and the pumps and plumbing required for circulating water and debris. The fuel assembly is consistent with the design described in DCD Section 4.2.2.2. In RAI-SRP6.2.2-SRSB-26, the staff asked the applicant to justify why the test results from the scaled-length, isothermal fuel assembly test facility are applicable to the post-LOCA long-term cooling situation for a full-length assembly in which boiling occurs in the upper portion of the fuel assembly. In its February 26, 2010, response, the applicant stated that most of the AP1000 fuel assembly debris head loss tests were performed with a single fuel assembly and upflow of water. The applicant included a fuel bottom nozzle, p-grid, and spacer grids in the single fuel assembly, simulating the bottom portion of a fuel assembly. The results from the fuel assembly tests have shown that the debris-induced pressure drop (dP) acceptance criteria are met. The upflow fuel assembly tests simulated DEDVI or DECL LOCAs where the debris entered into the downcomer and the core inlet region.

The fuel assembly test results show that the vast majority of the dP was seen across the inlet nozzle and p-grid in the bottom part of the test assembly; therefore, the applicant concluded that the use of a scaled-length test assembly will not change the test results because most of the pressure drop occurs in the first part of the test assembly. On the basis of the test results that show the majority of the dP occurs in the inlet nozzle and p-grid in the bottom part of the test assembly, the staff agrees that the scaled-length test fuel assembly is reasonable to simulate the AP1000 fuel assembly for debris blockage at the core inlet.

The applicant set up a test loop, as described in Appendix C to WCAP-17028, and configured it in two ways: (1) to simulate a DVI break or CL break with break flow through the downcomer and up through the core inlet region; and (2) to simulate a HL break with downflow from the break to the core exit region. The test facility also simulated additional conditions, such as water at higher temperatures and the boiling environment seen in the AP1000 core when there is a high level of debris blockage at the core inlet. The boiling environment in the core is simulated by injection of air during the test. The test loop flow rates were scaled to expected flows seen in the AP1000 to represent the flow rate at the core inlet for a DECL/DVI break with upflow. The selected flow rates were representative of or bounded flows expected through the core during the recirculation phase after a LOCA.

In response to RAI-SRP6.2.2-SRSB-26, the applicant performed additional fuel assembly Tests 36, 37, and 39 to address the applicability of the isothermal scaled test to potential boiling conditions at the full-length fuel assembly. The staff asked whether there are situations where two-phase flow could challenge the single-phase test results and whether a higher liquid temperature or local boiling phenomenon could affect the behavior of the debris plugging the core. Tests 36 and 37 repeated Test 30, with the exception that chemicals were added to the test loop to simulate reactor coolant chemistry and the test loop was heated to a higher temperature than for Test 30. The applicant's test results showed, as before, that the fiber accumulated around the bottom of the p-grid before accumulating around the bottom nozzle. For both tests, near the end of the flow sweeps, the debris bed was very thick and very smooth, yet it did not cause an increase in the steady-state core dP. The pressure drop increase occurred almost exclusively at the bottom nozzle/p-grid location as in the other tests. On the basis of its review, the staff agreed that using the lower temperature water in the test facility in Test 30 was a conservative effect on the measured core inlet dP. As a result, the staff found that testing with lower temperature water was acceptable.

The applicant performed an additional Test 39 in response to RAI-SRP6.2.2-SRSB-26 to observe the effect of adding debris to the test fuel assembly under simulated DEHL break conditions. The conditions simulated in this test were upward flow of coolant with boiling. The debris loadings used were the same as for Test 30. [

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In the WCOBRA/TRAC analysis, the applicant predicted significant two-phase mixture at the top of the AP1000 core. In APP-GLR-PXS-001, Sensitivity Case 10 was determined to be the worst case debris blockage case simulated by WCOBRA/TRAC. The applicant found that the pressure drop across the core for single-phase flow at the core inlet was larger than the pressure drop across the top part of the core even when two-phase flow at the top of the core causes the flow losses to be much greater.

On the basis of its review, the staff finds that the fuel assembly debris loading head loss testing using single-phase flow appropriately estimates the worst case dP across the core inlet when compared to the two-phase flow predicted in an AP1000 core after a LOCA.

Implicit in the single fuel assembly debris load head loss test is a basic assumption of uniform blockage across the core inlet. In RAI-SRP6.2.2-SRSB-33, the staff requested that the applicant discuss potential effects of non-uniform blockage in the fuel assembly test facility. In its response, dated January 29, 2010, the applicant explained that the AP1000 fuel assembly debris-loading head loss tests indicate that there is non-uniform flow blockage and there is considerable variation in the fuel assembly flow/dP even with the same debris addition. Every test has shown non-uniform blockages within the fuel assembly and gaps in the debris bed. This shows that the debris bed at one fuel assembly will be different from that in others.

The applicant further stated that it is expected that the distribution of debris across the core inlet of all fuel assemblies will be non-uniform. Debris that accumulates at the core inlet is not expected to distribute uniformly across the entire core inlet and, therefore, some fuel assemblies will have more debris buildup than other fuel assemblies. The fuel assemblies with less debris buildup will experience less dP across the fuel assembly and will be able to pass more flow through those fuel assemblies. The higher flows through these low dP fuel assemblies would cross over to assist cooling fuel assemblies that have higher dP and lower flow at the core inlet and provide additional margin for long-term cooling. Therefore, the staff concludes that the single fuel assembly debris-induced head loss test is conservative and acceptable.

Debris Addition Scheme

The manner in which the debris is added has been observed in prior industry testing programs to make a difference in the overall head loss through the fuel assembly. Therefore, it was determined that the order in which particulate, fiber, and chemicals were added was important. the applicant used the following debris addition schemes (i.e., sequential addition and concurrent addition) for the fuel assembly debris loading head loss tests:

- For tests applying the sequential additions of particulate, fiber, and chemicals, an aliquot of the water from the mixing tank was removed and placed in a container for each addition of particulate. The particulate was then well mixed into this water until completely suspended before being added to the mixing tank. The mixing tank volume was allowed to thoroughly mix. Then the fiber was added per the test plan using a similar manner as described above for the particulate. Thirdly, the surrogates for chemical reaction products were added to the test loop after all of the fiber and particulate had been added. The chemical precipitate was mixed outside the test loop per the WCAP-16530-NP-A methodology and then added to the test loop in measured batches. The approach of sequential debris addition into the test loop is consistent with the NRC-approved guidance on head loss testing, WCAP-16406-P-A, Revision 1. The fuel assembly tests using sequential debris addition were set up to determine the maximum dP value that would be achieved under the test condition.

- Tests applying the concurrent addition of particulate, fiber, and chemicals modeled debris addition at start of recirculation after a LOCA. For tests applying concurrent debris addition, an aliquot of the water from the mixing tank was removed and placed in a container and the prescribed amounts of particulate and fiber were added to the container. The particulate and fiber were then well mixed in this water until completely suspended before being added to the mixing tank. Concurrent with the addition of particulate and fiber, chemical surrogate was added directly to the mixing tank in the amount prescribed in the test procedure. Concurrent debris additions were made at times specified in the test procedure.

Test Acceptance Criteria

In the long-term cooling sensitivity studies described above, Case 10 provided the limiting case conditions, which were postulated assuming that all the debris will reach the core entrance after one complete recirculation of the entire containment water volume. For Case 10, the long-term core cooling analysis found that for a CL or DVI break, the limiting dP through the core is 28.3 kPa (4.1 psid) with a corresponding minimum core flow rate of 29.5 kg/s (65.0 lb /s or 480.7 gallons per minute (gpm) core flow rate), or [ ] per fuel assembly. Therefore, the acceptance criterion for the fuel assembly test is 28.3 kPa (4.1 psid) at [ ] per fuel assembly. This is identified as acceptance criterion 1 for all fuel assembly tests not involving a HL break.

The second criterion is based on sensitivity Case 3 in the long-term cooling analysis. In this case, the maximum dP allowed in the core at the beginning of the recirculation phase is 24.1 kPa (3.5 psi) with a flow rate of [ ] per fuel assembly. The second criterion is intended to verify that during the recirculation, for any level of the decay heat between the start of the recirculation phase (long-term cooling Sensitivity Case 3) and 8.6 hours, the core will be satisfactorily cooled, requiring that the head loss through the core at [ ] per fuel assembly must be lower than 24.1 kPa (3.5 psi). It should be noted that the concurrent debris addition fuel assembly tests were set up in order to model the plant timing of debris addition after the start of recirculation. Therefore, the test time can be related to the plant time. In contrast, the sequential debris addition tests were set up to determine the maximum dP value that would be achieved under the test condition, with no reference to the time at which it would occur in the plant. For this reason, the second acceptance criterion is not applicable to the sequential debris addition tests but only to the concurrent debris addition tests.

For those fuel assembly debris head loss tests where the debris is added in a mechanistic, time-dependent sequence for the beginning of recirculation (2.6 hours) after a DECL or DEDVI LOCA, the applicant used Sensitivity Case 3 for the core dP acceptance criteria. This case is conservative because: (1) debris-induced resistance at the core inlet is assumed to occur at the beginning of recirculation, earlier than the time debris would be transported to the core inlet; and (2) the decay heat rate at 2.6 hours post-LOCA is higher than the decay heat rate at 9.0 hours, the time at which the Sensitivity Case 10 acceptance criteria are used.

For a HL break the flow in the upper part of the core is expected to oscillate when the downflow in a low-power assembly becomes low and the steam generation rises upwards. Case 11 from the sensitivity studies found that the limiting dP through the core is 13.8 kPa (2 psid) with a corresponding minimum flow rate of 97.3 kg/s (214.5 lbm/s or 1,602 gpm core flow rate). This is the minimum flow rate that can be maintained for long-term core cooling. The acceptance criterion for the HL break fuel assembly tests is that the dP across the fuel assembly test article must not exceed 13.8 kPa (2 psid) at a flow rate of [ ]. This acceptance criterion will be applied to each of the HL break fuel assembly tests.

As discussed earlier, there are two acceptance criteria for the upflow fuel assembly tests simulating CL and DVI line breaks. The first acceptance criterion for the fuel assembly tests is that dP across the fuel assembly inlet must not exceed 4.1 psid at a flow rate of [ ], which is based on the limiting Case 10 of the long-term cooling sensitivity studies. This acceptance criterion was applied to each of the fuel assembly tests. The second criterion, applied to all concurrent debris addition tests, is intended to verify that during containment recirculation when the debris bed is not completely formed, for any decay heat rate between the start of the recirculation phase at 2.6 hours (long-term cooling Sensitivity Case 3) and the time of 8.6 hours that it takes to recirculate one containment water volume, the core will be adequately cooled, with the core head loss lower than 24.1 kPa (3.5 psid) at fuel assembly flow rates of [ ]. Another test acceptance criterion for a HL break is 13.8 kPa (2 psid) at a flow rate of [ ] through the fuel assembly.

The applicant conducted many CL tests with a constant flow rate higher than [ ] or with variable flow rates where the minimum flows were higher than [ ]. Moreover, in all the concurrent addition tests, the flow rate at a time before 9 hours in plant time was higher than [ ]. To evaluate the test results, the experimental results need to be adjusted to a lower flow rate ([ ] for the first criterion, [ ] for the second one). The applicant developed Equations 5.1.2 and 5.1.3 in WCAP-17028 to calculate the adjusted dP to flow rates of [ ] and [ ], respectively, for the comparison with the first and second acceptance criteria. These equations were developed based on the dP/flow rate relationship developed for each fuel assembly test described in Section 5.2 of WCAP-17028. The exponent "b" in Equations 5.1.2 and 5.1.3 is determined from the fuel assembly head loss tests and summarized in Table 5-1 of WCAP-17028 for each test. In Section 8.39 of WCAP-17028, the applicant described the development of Equations 5.1.2 and 5.1.3, as well as the value of exponent b in these equations.

In RAI-SRP6.2.2-SRSB-27 and RAI-SRP6.2.2-SRSB-29, the staff asked for information about the development of the b exponent in extrapolating the expected pressure differential and the use of the two acceptance criteria for the testing program. In its February 26, 2010 response, the applicant explained that many of the AP1000 tests were conducted with variable flow rates where the flow was changed during the test as the dP increased to simulate the actual behavior of the plant. Therefore, the applicant developed two acceptance criteria for the fuel assembly debris load tests. This was discussed in WCAP-17028 in Sections 5 and 8.

As discussed above, to determine whether a test met the long-term cooling criteria, the maximum dP measured in each test was adjusted, as described below, based on both the flow that existed when the maximum dP occurred and the minimum acceptance flow (i.e., [ ]). This adjusted dP provides a simple way to compare tests and determine whether the test met the acceptance criteria.

The data collected by the tests resulted in the development of Equation 8.39.1 in WCAP-17028, which defines the relationship between head loss and the flow rate. This equation is based on the Darcy formula, and the exponent is determined by test results. When the debris bed is formed and stable, the pressure drop behavior of the debris bed will vary consistently with flow rate. In other words, [ ]. This also means that once the value of the [ ] are known at a particular flow rate, it is possible to evaluate the value of dP at any flow rate.

Since the tests were not conducted at a low flow rate of [ ] as required by the first acceptance criterion, the measured dP must be adjusted as suggested above. For the first acceptance criterion, some of the tests performed flow sweeps to the low flow level. From each of these tests, an exponent was developed and used in Equation 8.39.4 of WCAP-17028 to calculate the pressure differential adjusted to the low flow of [ ]. To illustrate this process, Test 34 will be reviewed below.

Test 34 was performed to investigate the nature of the dP/flow relationship throughout the test to allow comparison of the bed behavior for a fully formed and stable debris bed. Flow sweeps were performed throughout the duration of Test 34, and the experimental results confirm that the dP and the flow are related by a power law relationship as shown in Equation 8.39.1 of WCAP-17028 even in the case of a debris bed not yet fully formed. [ ], which was much lower than the first acceptance criterion of 28.3 kPa (4.1 psid). The test dP at the maximum debris bed resistance as presented in Table 8-1 was used for the first acceptance criterion. Since Test 34 was a concurrent test, the second acceptance criterion must also be applied.

The second criterion is based on long-term cooling Sensitivity Case 3, which assumes the maximum blockage condition in the core at the beginning of the recirculation.

Concerning the applicability of Equation 8.39.3 to the second acceptance criterion, the use of the stable bed exponent would result in a greater underestimation of the adjusted dP. Looking at the results of Test 34, the exponent at [ ], and the bed resistance is still increasing. This suggests that a reduction of about 27 percent should be applied at the stable bed exponent to extrapolate the test results at higher flow rates down to [ ], which is the flow basis for the second criterion. For conservatism, a reduction of [ ] was applied to the exponent b of the second criterion. This reduction is based on the difference between the fully formed bed exponent and the lowest exponent estimated in Test 34. For the second acceptance criterion, the exponent b becomes [ ] because of the reduction applied to the exponent. Applying the second criterion to Test 34, the b exponent for Test 34 was determined to be [ ]; applying Equation 8.39.5 with a measured maximum dP at [ ], which was much lower than the second acceptance criterion of 24.1 kPa (3.5 psid).

For most tests, the value of the exponent applied to flow was determined by the flow/dP data taken from that test. For Tests 18 through 34 and Tests 36 and 37, flow sweeps were conducted at the end of the test that provided many data points (different flows/dPs). The flow sweeps reduced the flow rate to the acceptance criteria flow so that these data are directly applicable. Such data are also available for Tests 8 through 11 because of the use of oscillating flows during the tests. Tests 1 through 6 and 13 through 16 used the average exponent from all tests. Table 9-1 and Figure 9-1 of WCAP-17028 show these results. The data show considerable margin between the scaled dP and the acceptance criterion. For example, for Test Case 33, which has the highest scaled dP, there is close to 50 percent margin from the acceptance criteria.

In the AP1000, debris can transport into the RCS though the flooded HL and possibly into the upper parts of the core. In an actual HL LOCA in an AP1000, the flow in the upper part of the core is expected to oscillate over several minutes. When the downflow in a low-power assembly becomes low, the steam generation rises. The tests before Test 35 were all conducted simulating a CL break scenario and without simulating boiling. In RAI-SRP6.2.2-SRSB-31, the staff asked the applicant for more information and clarification concerning the HL break. In response to this RAI, the applicant performed more testing to simulate the effects of a HL break on the AP1000 core. These tests were conducted to prove that the CL breaks were more limiting even if debris entering the top of the core would occur in a HL break. These conditions were to be tested for an expected HL break LOCA in an AP1000.

In the event of a HL break, the applicant assumed that the post-LOCA containment debris load would be equal to 90.72 kg (200 lbs), consisting of 3.0 kg (6.6 lbs) of fiber, 87.72 kg (193.4 lbs) of particulates, and 25.85 kg (57 lbs) of chemical precipitates. The applicant assumed that all the debris was transported into the core through the outer ring of fuel assemblies.

Tests 35, 38, and 39 represented the HL breaks. The goal of Test 35 was to observe the effects of adding debris to a model fuel assembly under reverse flow conditions that might exist in an AP1000 reactor after a HL break. The test evaluated the distribution of debris blockages within the top portion of the assembly, and the relationship between pressure and flow. To model AP1000 behavior on a HL break, further testing of debris blockage with reverse flow was needed. With the debris load and chemical precipitate tested, the peak head loss recorded for this test was [ ]. The [ ] psid was below the acceptance criteria of [ ].

The purpose of Test 38 was to investigate the debris behavior in the outer fuel assemblies under simulated DEHL LOCA conditions, where flow could be downward and transport debris into the upper part of these fuel assemblies. The test also investigated the impact of changing the direction of flow from the downward to the upward direction, representing steam with boiling present in the upward direction. Test 38 was conducted to represent the downflow in a low-power assembly that produces steam and results in a change in flow from the downward to upward direction. This flow reversal and its effect on a debris bed were explored. The downflow was introduced for approximately 1 hour after one concurrent debris addition. The flow was then reversed to upflow, simulating the steam generation rise, and boiling was introduced at this time.

The reversal of flow did not visibly change the bed a great deal, but when air was introduced to simulate boiling, the [ ]. The air was injected at [ ], which is the maximum allowable air injection for the test rig and is much lower than the steam volume predicted to exist in the upper core region, and proved that this flow rate can quickly disperse any debris that would collect. Therefore any flow rate greater than [ ] would do the same, but more quickly. Based on the test results, the applicant concluded that the brief upflow of steam would be sufficient to break up any accumulated debris in the upper core region. The purpose of Test 39 was to investigate the debris behavior in central fuel assemblies that will be exposed to constant upward flow of water and steam. Test 39 was conducted solely to represent the HL break condition representing the steam in the upward flow direction and the local boiling phenomenon affecting the behavior of the debris plugging the core. The air was injected at the same rate in Rest 38 throughout the duration of the test. [ ].

Tests 38 and 39 had very low pressure drops, [ ].

[ ]. The pressure drops in the core for all three HL tests were less than the acceptance criterion of 13.8 kPa (2 psid).

In RAI-SRP6.2.2-SRSB-28 and RAI-SRP6.2.2-SRSB-30, the staff asked the applicant to explain the large variation in test results for Tests 27, 29, and 30, given they had the same amount of debris and debris addition procedures and appeared to be repeatable tests. The staff also noted that the fuel assembly test results indicate large uncertainties where the peak dP is significantly different for similar flow cases with the same amount of fiber. The staff asked the applicant to justify the acceptability of test results with large uncertainties and to provide an evaluation of the statistical confidence with which the test results could be used to assess the long-term cooling effectiveness based on the fuel assembly debris loading head loss tests.

In its response, the applicant prepared APP-GW-GLR-092, Revision 0, describing its statistical analysis of the fuel assembly debris loading head loss tests. The objective of the statistical analysis was to use available test data to show that there is a low probability that the AP1000 debris bed resistance will exceed the analyzed safety analysis limit from the long-term cooling sensitivity studies, which show acceptable results for the DVI break or CL break scenarios where the debris enters the core from the downcomer and lower plenum. The applicant concluded that, based on the test data and consistent with the statistical evaluation, it had established a conservative distribution of the adjusted pressure drop across the core. Using this conservative distribution, the effective adjusted pressure drop at the core inlet was calculated to be significantly below the safety analysis limit of 28.3 kPa (4.1 psid) with core flow of 29.5 kg/s (65 lbm/s).

The statistical analysis of the tests evaluated in APP-GW-GLR-092, Revision 0, determined that the probability for a single fuel assembly to exceed the acceptance criterion of 28.3 kPa (4.1 psi) is less than [ ]. Therefore, there is a low probability that a few of the 157 fuel assemblies in the AP1000 core could build up a debris bed that could exceed the acceptance criteria. However, many of the fuel assemblies in the core will have debris beds that have lower resistances than the acceptance criterion. The results of the statistical analysis of the AP1000 fuel assembly debris testing show that the effective core inlet adjusted dP would be [ ], using the 95 percent upper bound standard deviation, which demonstrates considerable margin to the acceptance criterion of 28.3 kPa (4.1 psi).

The staff notes that the statistical analysis is not required as a part of the GSI-191 evaluation, and the size of the test dataset is not sufficient to form the basis for a sound statistical analysis. However, the staff finds that the statistical analysis provides useful supporting evidence that the there is low probability that debris entering the core and debris bed buildup would degrade the core cooling margin to the point that the ECCS acceptance criteria were not met.

The staff evaluated the core cooling capability of the AP1000 core during long-term cooling based on the review of the long-term cooling analyses and fuel assembly debris loading head loss testing. The staff concludes the following:

- The design basis of 90 percent debris bypass (i.e., 90 percent of the design basis containment debris entering the core through submerged breaks unfiltered, bypassing the circulation screens), determined by the limiting DECL break, is a conservative assumption for the long-term cooling evaluation. The limiting DECL break was determined to be the worst case break that could allow the maximum amount of unfiltered debris to enter the reactor downcomer.

- The applicant conducted WCOBRA/TRAC long-term cooling sensitivity analyses using large nonmechanistic flow resistance at the core entrance to simulate debris-induced flow blockage to determine adequate core cooling. The limiting core inlet dP and flow results from Sensitivity Cases 3 and 10 were then used to develop acceptance criteria for the fuel assembly debris loading head loss testing.

- The applicant ran 39 fuel assembly tests to show that the worst case debris that would be expected in the AP1000 reactor during long-term cooling would not exceed the dP acceptance criteria. These tests showed that the design-basis AP1000 fibrous and particulate debris and chemical precipitates assumed to exist in the AP1000 containment do not induce a high enough head loss through the fuel assembly to reduce flow into the core to less than the minimum required to provide adequate long-term cooling following a LOCA.

On the basis of its review, the staff finds that the evaluations performed by the applicant showed that, with the design-basis containment debris loading, adequate core cooling in the AP1000 can be maintained during the post-LOCA recirculation long-term cooling period.

6.2.1.8.2.8 Head Loss and Vortexing

The applicant conducted head loss testing using plant-specific debris loads and flow rates to demonstrate the adequacy of the containment recirculation and IRWST screens. The debris types considered were particulate, fiber, and chemical precipitates. WCAP-16914-P and WCAP-16914-NP document the methodology, assumptions, and results. The staff evaluated these documents considering guidance from Enclosure 1, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," and Enclosure 3, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated March 28, 2008.

WCAP-16914-P documents all eight AP1000 screen head loss tests. These experiments were performed over a period of time, capturing different evolutions of the design. Because there were significant changes to the debris source term and screen size over the course of the program, only Tests WE213-4W and WE213-5W are representative of the final AP1000 design. These two tests, which are the focus of the subsequent discussion, were run with nearly identical parameters.

The AP1000 containment recirculation and IRWST screens comprise pockets constructed from perforated plate with holes less than or equal to 1.59 mm (0.0625 in). The screen used in the test is similarly constructed, [

].

The test facility included a large acrylic tank containing the test screen and supporting structure, positioned on the tank floor. Fluid entered the tank through a submerged sparger, located close to the upstream tank end. The flow entered the test screen pockets horizontally and was recirculated through piping connected to the downstream end of the tank. The pump, flow meter, and valves were external to the tank. The temperature of the fluid was maintained just below room temperature via cooling coils wrapped around the sparger.

Tests WE213-4W and WE213-5W began with clean screen head loss measurements at each of the two flow rates designated for use during the test to provide a reference point for the subsequent test results. The flume was then stabilized at the higher flow rate, and the particulate was added by slowly sprinkling the particulate surrogate at the water surface furthest upstream of the screen. The flume was run for five turnovers and stirred to resuspend any settled particulates. The fiber was then added in small batches. For each addition, a portion of the dry fiber was shaken into a solution of flume water and then slowly poured on the water surface furthest upstream of the screen. When fiber addition was complete, the flume was stirred and then allowed to run at a steady state overnight. The next morning, the flow rate was reduced to the minimum value in preparation for chemical addition. The chemical precipitates, created outside the loop, were slowly added to the flume in small batches, at a rate that bounded twice the predicted rate of chemical production in the AP1000, scaled to the limiting screen surface area. Upon completion, the flume water was stirred and again allowed to run at a steady state overnight. The next day, the flow rate was increased to the maximum value and swept back before ending the test. For both Test WE213-4W and test WE213-5W, the head loss remained very close to zero throughout the entire test, from the initial clean screen step through to the final flow sweeps.

The particulate debris comprises the particulate portion of latent debris and the coatings in the ZOI, which fail as fine particles. Coatings that fail as chips are not transported to the screen, as previously discussed. The particulate surrogate was [ ], which is consistent with the description of "dirt" from Appendix V to NEI 04-07. The surrogate for the latent fiber was [ ], which is consistent with the recommendations in Appendix VII to NEI 04-07. It comprised heat-treated fiber that had been either shredded or chipped. For Tests WE213-4W and WE213-5W, the fiber was submersed in a bucket of water and thoroughly shaken. It was not added to the flume until it was confirmed that all fibers were sufficiently fine and individualized. This approach eliminated the fiber agglomeration that was observed in some of the earlier tests. [ ] was used as a surrogate for all three chemical products generated in the postaccident AP1000 sump. As discussed in Section 6.2.1.8.2.4 of this report, this is an appropriate surrogate. The chemical reactant products were formed outside the test loop in accordance with the NRC-approved methodology from WCAP-16530-NP-A. The debris sequencing in Tests WE213-4W and WE213-5W is consistent with the guidance. The flume water was stirred for 1 minute after each debris type was added, which is expected to be sufficient time for all debris to initially reach the screen. Visual confirmation was not possible because the particulates, which were added first, turned the water gray.

For fiber and particulate debris, the loadings for the IRWST and containment recirculation screen are found by dividing the amount of debris transported to each screen by the screen frontal area. The results, based on values taken from the DCD, are presented in Table 5-2 of WCAP-16914-P along with the calculated debris loading from Tests WE213-4W and WE213-5W. As shown, the test debris loadings encompass both the IRWST and containment recirculation screens. Scaling the tested amounts of debris by the more limiting IRWST frontal areas and transport assumptions demonstrates the testing bounds a containment that has [ ] of fiber and [ ] of particulate. The AP1000 design has 3.0 kg (6.6 lb) of fiber and 87.8 kg (193.4 lb) of particulates; thus, there is significant margin in the tested amount of particulates.

A different approach was taken to determine the amount of chemicals added to the test flume. The intended amount, shown in Table 5-2 of WCAP-16914-P, was determined by scaling the chemical debris to screen frontal areas. The applicant claimed this amount of chemicals would result in a chemical concentration in the test flume about [ ] times higher than expected in the AP1000 containment. In order to remove some of this conservatism, the applicant added a termination criterion to the test plan that would allow the test to conclude if the measured [ ]. For both Test WE213-4W and test WE213-5W, the flume was sampled after a portion of the chemical precipitates was added; [

].

The applicant provided additional information about the concentration measurement procedures in response to RAI-SRP6.2.2-CIB1-26 and RAI-SRP6.2.2-SPCV-31. The staff did not conclude that the [ ] provided a conservative alternative to area-based scaling. However, based on the amount of chemical debris added and the measured head loss, the staff did conclude that the test conformed to the staff's guidance with respect to the amount of chemical debris. This is discussed below with the test results.

The procedures for Tests WE213-4W and WE213-5W included minimum and maximum flow rates, originally meant to differentiate the higher IRWST injection flows from the lower recirculation flows. The fiber and particulates were added at the maximum flow rate, representative of injection flow, and the chemicals, which would not precipitate until after start of recirculation, were added at the minimum flow rate. Tests WE213-4W and WE213-5W specified the same maximum flow rate, but the minimum flow rate in Test WE213-5W was slightly higher than in Test WE213-4W in order to encompass the recirculation flow oscillations observed in the long-term cooling analysis, APP-PXS-GLR-001. The highest flows in the AP1000 occur during operation of the nonsafety-related RNS system, which was not intended to be bound in the original test plan. However, the pressure drops remained near zero throughout the testing, indicating a clean screen. This is consistent with photographs from WCAP-16914-P, which clearly show open areas on the screen. Additionally, the calculated fiber bed thickness for these tests is [ ], which is [ ] times thinner than what was demonstrated to produce a thin bed in one of the earlier tests.

As stated above, the staff determined that, although not all of the chemical debris was added to the test, the test conformed to the staff's guidance for test termination with respect to chemical debris. In Section 16 of the March 2008 guidance for GSI-191 chemical effects, the staff stated that tests should be terminated in a way that demonstrates with high confidence that additional time or chemical debris would not significantly change the maximum head loss. In Tests WE213-4W and WE213-5W, the applicant added [ ] percent of the prepared chemical debris, which was scaled to screen frontal area (Table 6-1). The screen area was clean and there was no measured head loss even after adding a substantial fraction of the debris. The staff concluded that the test termination conformed to the staff's guidance.

Because the pressure drop remained negligible during the flow sweeps, the applicant was able to demonstrate that the screen performance was not a function of flow rate and the DCD criterion was changed to require that the testing encompass RNS operation. The test report incorporated this change in Table 5-2, which identifies maximum AP1000 flow rates as those associated with RNS operation. As shown, the maximum tested flow rate bounds the containment recirculation screen, but it is only 94 percent of the IRWST scaled value. The applicant considers this acceptable because a 6-percent difference is small, the scaling was conservatively based on frontal area rather than surface area, and the test results demonstrate a clean screen. The staff agrees that the conservatism in the scaling bounds the minor difference in flow rates and that the resultant clean screen will be insensitive to a 6-percent higher flow rate.

The DCD-specified head loss limit of 1.7 kPa (0.25 psia) was derived from Sensitivity Case 3 of APP-PXS-GLR-001. In APP-PXS-GLR-001, Table 4-1, the containment recirculation screen flow was modeled by the PXS A flow at a temperature of 93.3 °C (200 °F), and the IRWST screen flow was modeled by the PXS B flow at a temperature of 60.6 °C (141 °F). The calculated head loss for the containment recirculation screen was [ ] at a flow rate over the frontal area of [ ], and the calculated head loss for the IRWST screen was [ ] at a flow rate over the frontal area of [ ]. The long-term cooling analysis modeled the pressure drop as proportional to the value of flow squared. Applying this relationship at the minimum test flow from Test WE213-4W gave a pressure drop limit of [ ] at the containment recirculation screen and [ ] at the IRWST screen. The test plan conservatively set the head loss limit to 1.7 kPa (0.25 psia) at all flow rates, which is more restrictive than the calculated value for either screen. The head loss measured throughout the test was negligible, which demonstrates additional margin.

The flume water level was set about [ ] above the top of the screen, bounding both the containment recirculation screens, which have several feet of submergence, and the IRWST screens, which have a minimum submergence of 7.1 cm (2.8 in). As stated in Section 7.3 of WCAP-16914-P, no vortex formation or air entrainment into the screens was observed during any of the testing. [ ]. This experiment is considered bounding as the current design demonstrates a clean screen. The relatively small submergence of the IRWST screens could lead to concerns regarding flashing and deaeration if the pressure drop across the screen approached the head loss limit. However, because the testing demonstrated a clean screen and no vortexing was seen, even during the flow sweeps that encompass RNS operation, there is no concern of voiding or potential entrainment of vapor if potentially saturated liquid passes through the screens. There is also no concern that additional water could be held up in the IRWST as the water level drops below the screen during the wall-to-wall flooding case described in DCD Section 15.6.5.4C.3.

6.2.1.8.2.9 Downstream Effects—In Vessel

During the post-LOCA containment recirculation long-term core cooling phase, the containment debris and chemicals enter the reactor vessel, causing potential core flow blockage, boron precipitation, and plateout of chemical precipitates on the fuel cladding, resulting in degradation of core heat transfer. Section 6 of TR-26, presents an in-vessel evaluation, which assesses the impact of debris in the post-LOCA recirculating water on components inside of the reactor vessel, including the core inlet and fuel assemblies.

The effects of debris in the post-LOCA recirculating water flowing through the fuel assemblies were evaluated through the long-term cooling sensitivity analysis and the fuel assembly head loss testing, as discussed in Section 6.2.1.8.2.7 of this SE. This section provides an evaluation of the effect of potential chemical deposition and scale buildup on the fuel rods on maintaining effective heat transfer from the fuel rods to the coolant.

During post-LOCA long-term cooling, the AP1000 PXS design uses ADS-4 valves connected to the HLs to vent steam and a considerable amount of water from the RCS. The water that leaves through the ADS-4 valves carries boron and other chemicals out of the RCS, which automatically and effectively limits the buildup of these chemicals in the core. Therefore, boron precipitation in the reactor vessel is prevented by sufficient flow of PXS water through the ADS-4 valves to limit the increase in boron concentration of the water remaining in the reactor vessel. DCD Section 15.6.5.4C.4 describes an evaluation of post-LOCA long-term core cooling, as well as core boron concentration and boron precipitation. The analysis results indicated that the ADS-4 venting quality at the initiation of recirculation is about 50 percent and decreases to less than 10 percent at the time of wall-to-wall flooding. At the maximum vent quality, the maximum boron concentration peaks at about 7,400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS-4 vent quality decreases, reaching 5,000 ppm about 9 hours after the accident. The maximum boron solubility temperature is 14.4 °C (58 °F) at 7,400 ppm, which is virtually unattainable in the reactor vessel or the ADS-4 vent pipe. For the containment floodup case, the minimum sump injection head is adequate to maintain core cooling and limit boron concentration. The results show that venting of steam and water ensures that there is adequate liquid flow through the core to cool it and to prevent boron precipitation. Section 15.2.7 of NUREG-1793 discussed the staff's evaluation of the boron precipitation and concluded that: (1) the core remains cooled for the duration of the long-term cooling phase; (2) the boron concentration in the core keeps the core noncritical; and (3) boron precipitation will not occur to obstruct core coolant flow.

However, the long-term cooling analysis presented in the DCD was performed without consideration of containment debris. Because of the flow conditions of the core where there is a significant blockage at the core inlet created by unfiltered debris through the break, The applicant performed a long-term core cooling sensitivity study to evaluate the effects of core inlet blockage. This sensitivity study is described in APP-PXS-GLR-001, Revision 4.

The AP1000 long-term cooling sensitivity study results show that as the core inlet flow resistance increases, the core flow rate decreases, the quality of flow discharged through ADS-4 valves increases, and the boric acid concentration increases. For Sensitivity Case 10, which has the highest core inlet flow resistance, the core flow rate is predicted to be 65 lbm/s with a bounding pressure drop of 4.1 psid, the ADS-4 discharge flow quality is 0.49, and the maximum concentration of boron is 6,100 ppm in the core. This is less than the maximum concentration of 7,400 ppm in the DCD evaluation. Therefore, there is no concern with precipitation in the lower plenum during long-term cooling following a LOCA since the reactor water is not cooled to temperatures lower than the corresponding boron solubility temperature during long-term cooling.

In RAI-SRP6.2.2-SRSB-26, the staff asked if there were any situations where two-phase flow behavior could challenge the single-phase fuel assembly debris-induced head loss test results, and whether a different liquid temperature or local boiling phenomenon affects the behavior of the debris plugging the core. In partial response to RAI-SRP6.2.2-SRSB-26, the applicant provided TR APP-GW-GLR-110, Revision 0, which provides an evaluation of the potential for plateout of unbuffered boric acid or buffered boric acid on the fuel rod surface. The applicant addressed the effect of boron plateout on the core for high steam qualities in the core region that may occur during a significant core inlet blockage by debris. The applicant had previously conducted a series of single-rod bench-scale tests to investigate the nucleate boiling heat transfer characteristics of unbuffered and buffered boric acid solutions. These tests included concentrations of boric acid and buffer agent TSP that were equal to and greater than those concentrations that will occur in the AP1000 following a LOCA. The applicant also reviewed additional PWR heated rod testing in the presence of boric acid solution with decay heat level heat input and low pressure for application to the AP1000. These additional heated rod tests included rod-bundle geometries (Tuunanen, J., et al., "Experimental and Analytical Studies of Boric Acid Concentrations in a VVER-440 Reactor during the Long-Term Cooling Period of Loss of Coolant Accidents," *Nuclear Engineering and Design*, issued 1994) and multirod full-height slab core geometry (W3F1-2005-0007, "Supplement to Amendment Request NPF-38-249, Extended Power Uprate," dated February 5, 2005). These more prototypical geometries of the multirod and rod bundle test displayed precipitation behavior in the heated rod region that was consistent with the single heated rod testing for unbuffered boric acid and boric acid buffered with TSP. These tests generally showed no bulk precipitation in the heated rod region of the core and some local precipitation in the boiling region if the core would become uncovered. The form of boric acid precipitation is usually amorphous and can be redissolved in the presence of a continuous liquid phase. Therefore, the applicant does not expect the deposition of boric acid or boric acid buffered with TSP to occur during post-LOCA conditions in the AP1000 since the heated core region has been shown to be covered at all times by a two-phase mixture.

For the AP1000 design, the boron concentration calculated for the limiting long-term cooling Sensitivity Case 10 is only 6,100 ppm, which has a corresponding low solubility temperature that is not attainable during the long-term core cooling period. Since the available test data demonstrate that, if the boron precipitation occurs at high boron concentration, boron precipitation on the fuel is generally amorphous and would be redissolved in the presence of continuous liquid phase, the staff agrees that potential boric acid solute plateout on the fuel cladding is not likely, and boron precipitation is not a concern for the AP1000 design.

To address the concern that post-LOCA containment debris and chemical precipitates can plate out on fuel rod cladding and impede the heat removal from the fuel rods, The applicant evaluated the impact of post-LOCA deposition of chemical precipitates on fuel rods for the AP1000 design. The source of chemical products is the interaction of the fluid inventory in the post-LOCA containment sump environment with debris and other materials exposed to and submerged in the sump fluid. The purpose of the evaluation is to predict a maximum scale thickness of the resulting cladding deposit buildup and the maximum clad/oxide interface temperature resulting from the deposits during the post-LOCA recirculation long-term core cooling phase.

The evaluation, described in Section 6.2.2 of TR-26, Revision 8, uses the Loss-of-Coolant Accident Deposition Model (LOCADM). LOCADM is a spreadsheet calculation tool developed by the PWR Owners Group to conservatively predict chemical interaction precipitate formation and buildup of chemical deposits on fuel cladding after a LOCA. The details of the LOCADM analysis model appear in Section 7 and Appendix E of WCAP-16793-NP, “Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid,” Revision 1, issued April 2009.

WCAP-16793-NP, Revision 1, also proposes the following two acceptance criteria for the post-LOCA evaluation of the chemical deposition on the fuel rods, which the staff accepted for GSI-191 consideration:

(1) The maximum cladding temperature during recirculation from the containment sump will not exceed 426.7 °C (800 °F).

(2) The thickness of the deposition of debris, chemical precipitates, or both on the cladding (oxide + crud + precipitate) will not exceed 50 mils (1270 Micrometer (μm)).

<u>Evaluation of the LOCADM Analysis Model</u>

LOCADM predicts both the deposit thickness and cladding surface temperature as a function of time at a number of core locations. Although the LOCADM analysis model is described in WCAP-16793-NP, Revision 1, there is a link to WCAP-16530-NP-A since the chemical model contained in TR WCAP-16530-NP-A is used to develop the potential source term of species that may enter the reactor vessel. WCAP-16530-NP-A provides a method for evaluating plant-specific chemical effects in a post-LOCA environment, including guidance for how to prepare surrogate chemical precipitates that may be used in strainer head loss tests. The staff reviewed and approved WCAP-16530-NP. WCAP-16530-NP-A, however, does not explicitly address potential chemical effects that may occur in the reactor vessel. WCAP-16793-NP, Revision 1, evaluates potential chemical effects that may occur downstream of the sump strainer in the reactor vessel. The materials tested in WCAP-16530-NP included:

1. commercially pure aluminum and galvanized steel,
2. calcium silicate (Cal-Sil) insulation,
3. NUKON™ fiberglass,
4. other fiberglass - Temp Mat™,
5. Interam™ E-class insulation,
6. powdered concrete,
7. mineral wool insulation,
8. microporous insulation (e.g., Min-K™), and
9. fire-retardant material (e.g., FiberFrax™).

WCAP-16530-NP describes a number of dissolution tests conducted to examine the chemical behavior of various materials found in the sump environment. Sampling times for the dissolution test were set at 30 minutes, 60 minutes, and 90 minutes. The results of the WCAP-16530 test program are consistent with previous work such as the integrated chemical effect test (ICET) program and show that:

1) The predominant materials leached from containment materials are:

    - aluminum ions
    - silicates
    - calcium ions

2) The predominant chemical precipitates formed are:

    - aluminum (oxy) hydroxide
    - sodium aluminum silicate
    - calcium phosphate (for plants using TSP for pH control)

It is possible that other silicate materials may be generated (e.g., calcium aluminum silicate or zinc silicate), but their contribution, based on the referenced studies, will be small (contributing less than 5 percent of the total mass) relative to the predominant precipitates.

The WCAP-16530-NP model considers the release rates of aluminum, calcium and silicate, as these provide the greatest masses of materials that can become insoluble and impacts of other materials are negligible. Given a source term of material from the WCAP-16530-NP model, the staff reviewed the methodology used to determine that these materials:

1. Would not deposit on fuel surfaces to the extent that heat transfer is unacceptably low, and

2. Would not block flow through the fuel channels should the scale materials deposited become dislodged by spalling during fuel cool down.

In evaluating the potential for plate-out of dissolved or suspended chemical compounds on the fuel surface, the WCAP-16793-NP methodology assumes that all of the dissolved species and compounds resulting from the WCAP-16530-NP assessment are transported through the containment sump screen to the reactor vessel. This material represents the source term in WCAP-16793-NP for evaluating plate-out of scale-forming materials on the fuel cladding. The staff finds this source term assumption to be acceptable since the chemical source term is based on WCAP-16530-NP testing, and it is conservative for the reactor vessel fuel analysis to assume that no precipitate settles on the containment floor, no precipitate becomes trapped in a filtering debris bed covering the sump strainer, and material does not deposit in other locations downstream of the strainer (e.g., heat exchangers, reactor vessel lower plenum).

Although the staff finds the use of the chemical model spreadsheet in WCAP-16530-NP to be acceptable for determining the chemical source term for LOCADM, a limitation and condition was provided in the safety evaluation for WCAP-16530-NP related to the aluminum release rate. The WCAP-16530-NP chemical model aluminum release rate is based, in part, on a fit to ICET data using an averaged 30-day release. Actual corrosion of aluminum coupons during the ICET test appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP model under-predicts aluminum release by about a factor of 2 during the active corrosion part of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. To account for potentially greater amounts of aluminum during the initial days following a LOCA, a user's LOCADM input shall apply a 2x increase to the WCAP-16530 spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP spreadsheet, but the timing of the release is accelerated. Alternately, users may choose to use a different method for determining aluminum release, but users shall not use an aluminum release rate equation that under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1. If a user uses plant-specific refinements to reduce the chemical source term calculated by the WCAP-16530-NP base model, the user shall provide technical justification demonstrating that the refined chemical source term adequately bounds the postulated plant chemical product generation.

WCAP-16793-NP uses various heat transfer computer programs [ANSYS Mechanical Software and WCOBRA/TRAC WCAP-12945-P-A] and a commercially available calculational software package (MATHCAD) for estimating the effects of the plateout of dissolved materials on the increase in fuel clad temperature. WCAP-16793-NP relied on the LOCADM code for its final assessments since the LOCADM calculations address non-uniform chemical deposition due to variation of core power and boiling.

The starting assumption for the LOCADM model with respect to chemical effects is that all the dissolved and suspended chemicals pass through the containment sump screen and into the reactor core. This is a conservative assumption because it maximizes the amount of chemicals available to cause deleterious effects.

The LOCADM model also assumes that some of the fibrous material from destroyed insulation is not removed by the sump strainer and that this material also passes on to the reactor core area. The mass of fiber passing through the strainer is determined on a plant-specific basis, based on bypass testing. LOCADM assumes instantaneous chemical participation of this fiber. Therefore, the fiber bypass quantity is converted to a mass of fiberglass and then to an equivalent mass of elements [calcium + aluminum + silicon] that is immediately available to be deposited in the LOCADM analysis. This increase in the mass of dissolved chemicals is compared to the original mass of dissolved chemicals determined by the WCAP-16530-NP calculations [calcium + aluminum + silicon] and a percent increase is calculated. This increase is on the order of one to two percent, and is referred to as a “Bump-Up Factor.”

These two chemical sources are then used in the plant-specific application of LOCADM. Given the potential plant-specific chemical source term in the reactor vessel, LOCADM determines the amount of scale that deposits on the fuel over time and then calculates maximum fuel clad temperature. An assumption that is very important to the LOCADM calculations is the coefficient of thermal conductivity for the chemical deposits. In order to determine an appropriate thermal conductivity coefficient for the LOCADM calculations, two different thermodynamic equilibrium based codes were used to assess the chemical species that may form in the post-LOCA reactor vessel environment. The applicant performed an analysis using the HSC program by Outokomptu. This thermodynamic equilibrium code was used to evaluate potential differences in the predicted species and to support the choice of a limiting thermal conductivity value for a chemical deposit that may form on the fuel. Using the chemical species predicted by these thermodynamic equilibrium analyses, a lower-bound thermal conductivity value was selected for the LOCADM analysis in WCAP-16793-NP to minimize heat transfer and maximize the temperature rise on the fuel surfaces. A chemical deposit thermal conductivity value of 0.19 watts per meter-degree Celsius (W/m-ºC) (0.11 British thermal unit per foot-hour-degree Fahrenheit (BTU/ft-hr-°F)) was selected based on the possible formation of a postulated sodium aluminum silicate scale. A thermal conductivity value of 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) is the minimum thermal conductivity value reported for sodium aluminum silicate scale. For comparison, the thermal conductivity of dry fiberglass insulation is approximately 8.65×10^{-2} W/m-ºC (0.05 BTU/ft-hr-°F), and, with eight percent of its mass wetted, it increases to approximately 0.173 W/m-ºC (0.1 BTU/ft-hr-°F). The staff questioned if there were any materials from the thermodynamic predictions for fuel clad surface deposits which could have lower thermal conductivity values. The applicant responded that 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) was a bounding thermal conductivity value reported for any of the postulated species that could form a scale deposit on the fuel clad surface. Information provided by the applicant in RAI response number 34 [Schiffley, F. P., PWROG letter to Document Control Desk, NRC, “Response to the NRC for clarification to Requests for Additional Information (RAI) on WCAP-16793-NP,” showed thermal conductivity coefficients of representative calcium-based boiler scale deposits that were in the 0.52 to 0.865 W/m-ºC (0.3 to 0.5 BTU/ft-hr-°F) range, and the thermal conductivity of glass was reported as 1.02 W/m-ºC (0.59 BTU/ft-hr-°F).

Since the LOCADM calculations do not consider the presence of large debris, the staff questioned whether small pieces of insulation (“fines”) incorporated into a deposit could result in a lower thermal conductivity value than the 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) assumed for a sodium aluminum silicate scale. The applicant responded that since core temperatures have decreased by the time the ECCS switches from injection to recirculation mode, which is the time when the first fibrous debris could bypass the sump screens and enter the core, the temperature of the core is insufficient to cause melting of the fiberglass or other fibrous material. Therefore, the presence of fiber fines would not create a different type of scale other than that predicted by the thermodynamic models. The applicant also responded that although dry fiberglass has a lower thermal conductivity than the 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) assumed for the chemical deposit, a fiber deposit would be porous and would allow water to fill in the porosity. Since water has a much higher thermal conductivity than air, the overall thermal conductivity for a deposit containing fiberglass would be bounded by the assumed 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) value. This reasoning is supported by literature [Joint Departments of the

Army and Air Force, USA, Technical Manual TM 5-852-51/AFR 88-19, Volume 5, Article Sub-Arctic Construction: Utilities, Chapter 12] that indicated the fiberglass thermal conductivity constant increases by a factor of two with an eight percent volume of water incorporated into its structure. This is also consistent with insulation manufacturer recommendations to change insulation if it is wetted since the heat conduction through the insulation increases; in other words, it is no longer an effective insulator.

Based on the above discussion, the staff finds that the 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) thermal conductivity value assumed for deposition of scale and particulate represents an acceptably low value to help achieve a conservative prediction of fuel clad temperature increases due to chemical deposits. If plant-specific calculations use a less conservative thermal conductivity value for scale, i.e., greater than 0.19 W/m-ºC (0.11 BTU/ft-hr-°F), the staff expects the licensee to provide a technical justification for the plant-specific thermal conductivity to the staff. This justification should demonstrate why it is not possible to form a sodium aluminum silicate scale or other scales with conductivities below the selected plant value.

Given the potential chemical source term and using a conservative value for thermal conductivity, LOCADM calculates deposit growth over time. The default initial oxide and crud thicknesses assumed by LOCADM are based on the fuel age and the limiting values that have been measured at modern PWRs. Since the boiling deposition mechanism results in the most rapid deposit growth and forms the most tenacious deposits, LOCADM assumes that all deposition occurs through the boiling process if conditions at a core node predict any boiling. The amount of scale calculated to be deposited under boiling assumes that 50 percent of the water present at the clad surface boils and all solutes transported into the deposit by boiling are deposited locally, as liquid evaporates, at a rate proportional to the steaming rate. Subsequent plate-out of solids, once boiling subsides, is estimated from other literature sources (WCAP-16793-NP, Revision 1, RAI Set #2, RAI #8) to be 1/80th of the solids deposition rate during boiling based on the temperatures encountered at the fuel. Once formed, deposits are assumed not to thin by flow attrition, dissolution, or spalling. The sample LOCADM calculation in WCAP-16793-NP, Revision 1, included a 3188 megawatt-thermal PWR with high fiber (198 m^3 (7000 ft^3)) and a large quantity of calcium silicate insulation (2.27 m^3 (80 ft^3)). The staff questioned what additional effect the existing clad crud film and oxide scale (from three cycles) would have on the LOCADM calculations. The applicant responded that the sample LOCADM calculation, for the conditions stated above, including initial fuel clad oxide and crud, showed the maximum chemical scale thickness calculated over 30 days was 10 mils (0.010 in). The maximum clad surface temperature after the start of recirculation was 162.2 °C (324 °F), which meets the acceptance criteria of 426.7 °C (800 °F).

Since LOCADM does not directly account for fiber fines bypassing the sump screen, the staff also questioned how possible effects from fibers depositing in the core are assessed. Analysis of core inlet blockage is discussed elsewhere in this report, but modeling demonstrated that with 99 percent of the core flow blocked, sufficient cooling water would be provided as a result of boiling and back flow from above to prevent clad temperatures exceeding 426.7 °C (800 °F). To model potential local hot spots, heat transfer analysis was provided in Appendix D of WCAP-16793-NP, Revision 1, assuming heat transfer in the radial direction only (i.e., ignoring any axial heat transfer) and using a chemical scale thermal conductivity of 0.173 W/m-ºC (0.1 BTU/ft-hr-°F). These calculations showed that for a chemical scale thickness of 0.127 cm (50 mils or 0.050 in) that formed “instantaneously” at the start of recirculation, the maximum fuel clad surface temperature for a fuel rod diameter of 0.91 cm (0.36 in) is 293.3 °C (560 °F). Additional analyses were performed for larger diameter fuel rods, 1.06 cm (0.416 in) and 1.07 cm (0.422 in) OD rods. The predicted peak clad-oxide interface temperature was less than the acceptance basis value of 426.7 °C (800 °F) in each case. The staff finds this analysis to be acceptable since the assumptions of instantaneous chemical precipitate formation, heat transfer only in the horizontal plane (radial direction), and the assumed thermal conductivity for chemical scale are judged to be conservative for reasons stated in the WCAP.

The staff also questioned whether blockage of core flow channels might occur from scale initially deposited on the fuel surface that would flake off during the cool down process. The applicant responded that the thickness of the scale formed is limited by the amount of solids dissolved in the water. Using scale deposition models the applicant demonstrated that the thickest scale fragment would be insufficient to bridge a fuel rod to fuel rod span to block flow. The staff finds this justification acceptable because the spalling process from the fuel is slow, and experience from spent fuel pool debris generated at PWRs shows these scale materials to be granular and of small size rather than large flakes.

The staff reviewed the LOCADM analysis model as described in WCAP-16793-NP, Revision 1, and finds that:

1. The mass of material used to determine the debris and scale loading is conservative based on the source term calculated from the WCAP-16530-NP tests, along with the assumption that no precipitates settle on the containment floor, are filtered at the sump screen, or deposit in heat exchangers, piping, or in the reactor vessel outside of the core. The mass of materials includes a “bump up factor” to account for fibrous material that bypasses the sump screens. The staff finds this bump up factor to be acceptable for reasons stated in this SER section.

2. The thermal conductivity assumed for chemical scale and debris deposits represents an acceptably low value (0.19 W/m-ºC (0.11 BTU/ft-hr-°F) to help achieve a conservative prediction of fuel clad temperature increase. Wetted insulation allows for better conduction of heat and the thermal conductivity of wetted insulation would be higher. Thus the use of 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) is a conservative assumption.

3. Industry-recognized calculation models were used to predict temperature increases at the fuel surface as a result of chemical plate-out, and these models confirm that the limit of 426.7 °C (800 °F) is not exceeded when these models are used in conjunction with the source term assumptions in WCAP-16530.

4. Blockage of fuel rod spans by spalled fuel scales is unlikely due to the time dependency for spalling and the small thickness of the scale compared to the space between the fuel rods.

Based on the above, the staff concludes that the LOCADM analysis model provides a valid approach to determining potential flow restrictions due to chemical effects of RCS liquid and containment debris and materials, and is both conservative and representative of the post-LOCA conditions based on chemical reactions described in WCAP-16530-NP. Therefore, given the acceptance criteria for fiber bypass, the staff concludes the chemical effects on core cooling resulting from debris and scale deposition following a LOCA are insufficient to create a condition resulting in fuel clad temperatures exceeding the temperature limit of 426.7 °C (800 °F). However, the acceptability of the application of the LOCADM analysis model is contingent upon the following conditions.

1. The aluminum release rate equation used in WCAP-16530-NP provides a reasonable fit to the total aluminum release for the 30-day ICET tests but under-predicts the aluminum concentrations during the initial active corrosion portion of the test. To provide more appropriate levels of aluminum for the LOCADM analysis in the initial days following a LOCA, users shall apply a factor of two to the aluminum release rate as determined by the WCAP-16530-NP spreadsheet. If a user chooses to use a different method for determining the aluminum release, it must demonstrate that the method does not under-predict the aluminum concentrations measured during the initial 15 days of ICET 1.

2. If plant-specific refinements are made to the LOCADM base model to reduce conservatisms, the user shall demonstrate that the results still adequately bound chemical product generation. If a user uses plant-specific refinements to the WCAP-16530-NP-A base model that reduces the chemical source term considered in the downstream analysis, the user shall provide a technical justification that demonstrates that the refined chemical source term adequately bounds chemical product generation. This will provide the basis that the reactor vessel deposition calculations are also bounding.

WCAP-16793-NP, Revision 1, states that the material with the highest insulating value that could deposit from post-LOCA coolant impurities would be sodium aluminum silicate. The WCAP recommends that a thermal conductivity of 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) be used for the sodium aluminum silicate scale and for bounding calculations when there is uncertainty in the type of scale that may form.

To demonstrate acceptable AP1000 long-term core cooling performance, the applicant performed an evaluation using the LOCADM spreadsheet to account for chemical reactions within the coolant that could lead to deposition of material within the core. This evaluation was documented in Section 6.2.2 of TR-26 and Westinghouse calculation note APP-PXS-M3C-057, “Loss of Coolant Accident Deposition Model (LOCADM) Analysis for AP1000 Plant Design,” Revision 1, issued November 2009. The AP1000 LOCADM evaluation makes the following assumptions and simplifications:

- AP1000 Unique Design Features:

  In the AP1000 design, the containment spray system is locked out during a LOCA. Therefore, the containment spray is not considered in the calculation of the post-LOCA debris source release.

  The AP1000 plant design relies on the ADS-4 valves in the HL to vent significant quantities of water along with steam from the core to the containment throughout the LOCA event. This behavior is modeled in the LOCADM spreadsheet by defining core injection flow rates that exceeded the boiloff rate by an amount calculated with the decay heat.

- Treatment of Aluminum:

  Although the AP1000 design precludes a large amount of aluminum from making contact with post-LOCA containment fluids, a mass of 27.2 kg (60 lb) of aluminum is assumed for conservatism. The aluminum surface area is increased to account for the zinc release from galvanized steel, which is not an input for LOCADM. Increasing the aluminum surface area is conservative because the aluminum release rate is greater than that of any other material used in this evaluation.

The aluminum release rate is modified to satisfy NRC concerns about the trend of the predicted aluminum corrosion, by doubling the release rate during the initial portion of the event, yet it holds fixed the total aluminum mass release. This is consistent with the condition of WCAP-16530-NP. It is important because the release rate of aluminum is increased early in the transient when the deposition on the fuel is greatest because of high core decay heat rates and the boiling associated with the removal of that decay heat.

- Use of the Prefilled Reactor and Sump Option

  The LOCADM analysis assumes that the entire sump volume is present in the sump at time 0, precluding the need to specify individual break flow rates. This is conservative, as the entire sump volume is immediately available to react with the submerged debris at the start of the transient, and provides for the calculation of a greater amount of chemical precipitate deposition on the fuel.

- Core deposition is assumed to begin at the start of recirculation (9,300 seconds) and continue for the 30 days evaluated.

- Use of the Bump-Up Factor To Account for Fibrous Debris

  The bump-up factor used in LOCADM accounts for the postulated bypass of latent fibrous debris by increasing the mass of chemical precipitates that may be deposited on the fuel. To implement the bump-up factor in LOCADM, all materials that contribute to the formation of chemical precipitates are increased by a uniform percentage so that the resulting precipitates available for deposition have increased by approximately the amount of latent fibrous debris assumed for the AP1000. This method is independent of the type, diameter, or length of the fiber.

  The bump-up factor as applied to the AP1000 LOCADM evaluation is conservative because it is calculated (in APP-PXS-M3C-053, "AP1000 Latent Debris Calculation," Revision 2, issued November 2009) based on a fibrous debris loading of [ ]. This is the same debris loading used in the ex-vessel downstream effects analysis to evaluate the effects of debris in the recirculating water on pumps, valves, and other components in the post-LOCA recirculation flowpaths. This value is much higher than the design-basis containment residual fiber debris value of 3.0 kg (6.6 lb) and 90 percent transport to the reactor vessel. There are other conservatisms as discussed in TR-26, such as the low value of thermal conductivity (0.19 W/m-ºC (0.11 BTU/ft-hr-°F) assumed in LOCADM for the scale buildup on the fuel rods when considering the latent fiber as part of the fuel rod post-LOCA scale.

  TR-26 indicates that the thermal conductivity of manmade fibers such as nylon and polyester (0.249 and 0.225 W/m-ºC (0.144 and 0.13 BTU/ft-hr-°F) are higher than the assumed thermal conductivity for the scale. The thermal conductivity of natural fiber, such as cotton (3.46E-2 W/m-ºC (0.02 BTU/ft-hr-°F) may be lower, but it will increase significantly when saturated with water, as is the case in a post-LOCA environment. The thermal conductivity of these saturated fibers rises significantly, trending towards the value of water at the ambient conditions saturating the fibrous material (~0.69 W/m-ºC

(0.40 BTU/ft-hr-°F), which is much higher than the heat conductivity used for the chemical scale in the LOCADM evaluation.

TR-26 presented three scenarios evaluated with the LOCADM spreadsheet for the AP1000 design: (1) minimum sump volume case; (2) maximum sump volume case; and (3) minimum sump volume case with a bump-up factor for fiber deposition. The detailed evaluation described in Westinghouse calculation note APP-PXS-M3C-057, Revision 1, includes more sensitivity cases. The results of sensitivity analysis indicate that the minimum sump water volume results in a higher concentration of AP1000 accident chemical products and is therefore more limiting for the chemical deposition evaluation. All three cases described in TR-26 include the doubling of the aluminum release rate as recommended by WCAP-16793, Revision 1. The results show the post-LOCA scale thicknesses of [ ], respectively, for the three cases. With the preaccident oxide thickness of 0.15 mm (5.98 mils) and a crud thickness of 0.14 mm (5.51 mils), the total deposition thicknesses are [ ], respectively. These predicted thicknesses are significantly below the acceptance criterion of 1.27 mm (50 mils). The maximum temperature calculated for the outside diameter of the fuel cladding (at the fuel/oxide interface) is [ ] for all three cases, which is much less than the acceptance value of 426.7 °C (800 °F). This peak cladding temperature occurs at the onset of recirculation before significant debris deposition on the fuel cladding occurs. The chemical deposition appears to have an insignificant effect on the peak cladding temperature because the decay heat is decreasing faster than the chemical deposition rate.

In summary, the LOCADM calculations performed for the AP1000 demonstrate that both acceptance criteria for long-term core cooling identified previously in this report are achieved with significant margin. Specifically, the following is true for the cases evaluated:

- The maximum clad temperature calculated for the AP1000 of [ ] is significantly less than the acceptance value of 426.7 °C (800 °F).

- The total thickness of deposition calculated for the AP1000 fuel cladding is significantly less than the acceptance value for thickness of 0.127 cm (50 mils).

Therefore, the staff concludes that the AP1000 long-term core cooling capability remains viable in the presence of chemical deposition on the fuel cladding.

6.2.1.8.2.10 Debris Source Term

This section evaluates how the plant demonstrates and controls the debris source term. For the AP1000, this includes ITAAC, COL items, and technical specification (TS) surveillance requirements.

ITAAC related to the debris source term are included in DCD Tier 1, Table 2.2.3-4, as part of the 8c) design commitment that the PCS provide safety injection during design-basis events. ITAAC Item ix) verifies by inspection that insulation inside containment within the ZOI is MRI or that a report exists demonstrating that it is a suitable equivalent insulation. ITAAC Item ix) also verifies by inspection that other insulation inside the containment below the maximum DBA flood level is MRI, jacketed fiberglass, or a suitable equivalent insulation. Item x) verifies by inspection that reports exist concluding that tags and signs inside containment that are not inside cabinets or other enclosures have a density greater than or equal to 1.6 g/cm^3 (100 lbm/ft^3) and that ventilation filters and fiber barriers inside containment within the ZOI or below the maximum DBA flood level have a density greater or equal to 1.6 g/cm^3 (100 lbm/ft^3).

The staff finds these ITAAC acceptable because they capture key assumptions made in the long-term cooling analysis regarding debris transport and the amount of fibrous debris and because DCD Tier 2, Section 6.3.2.2.7.1, clearly defines the ZOI, the maximum DBA flood level, and the requirements for a suitably equivalent insulation.

COL Information Item 6.3-1 requires applicants referencing the AP1000 design to develop a cleanliness program that limits the debris left inside the containment following refueling and maintenance outages. Specifically, the amount of latent debris located within the containment must be less than 59.0 kg (130 lb) total latent debris, of which up to 3.0 kg (6.6 lb) is fibrous, and any outage materials stored inside the containment must not produce physical or chemical debris that could be transported to any of the filtering locations. The staff finds this acceptable because it is consistent with the recommendations in RG 1.82, Section C.1.1.2.1, which states that plant procedures should be established to regularly clean the containment and to control and remove foreign materials, and Section C.1.3.2.5, which states that the cleanliness program should be correlated to the amount of debris used in the long-term cooling analysis.

TS Surveillance Requirement 3.5.6.8 requires visual inspection of the IRWST and recirculation screens every 24 months to ensure that they are not restricted by debris. TS Surveillance Requirement 3.5.4.7 requires a similar 24-month inspection of the IRWST gutters. This is consistent with the long-term cooling analysis, which assumes the screens are clean before the LOCA.

6.2.1.8.2.11 Screen Design

ITAAC related to the screen design are included in DCD Tier 1, Table 2.2-3-4, as part of the 8c) design commitment that the PCS provide safety injection during design-basis events. Items vii) and viii) verify by inspection the key design features of the debris screens and barriers, including the existence of plates and debris curbs for the containment recirculation screen and the raised elevation of the IRWST screens. The screen frontal areas, surface areas, and mesh hole sizes are also verified to meet specific design criteria. Because these criteria are consistent with the head loss testing, they demonstrate that the as-built design will perform as expected.

Regulatory Position C.1.1.1.2 of RG 1.82, Revision 3 states that to the extent practical, the sumps should be physically separated from each other and from high energy piping systems by structural barriers to preclude damage by whipping pipes or high velocity jets of water or steam.

While there is physical separation between the IRWST screens, the applicant states that it was necessary to position the containment recirculation screens next to each other due to the location of the PXS subcompartment and the large size of the screens. To address pipe rupture and jet impingement vulnerabilities, the applicant has committed to demonstrate that the containment recirculation screens are protected by pipe whip restraints from the dynamic effects of pipe breaks in DCD Tier 2 Section 3.6.4.1 and DCD Tier 1 Table 3.3-6, Item 8). The staff finds that these restraints will protect the containment recirculation screens such that there are no credible pipe ruptures or jet impingement scenarios capable of causing screen failure. Therefore, this design meets the requirements of GDC 35.

Regulatory Position C.1.1.1.1 of RG 1.82, Revision 3 states that a minimum of two sumps should be provided, each with sufficient capacity to serve one of the redundant emergency core cooling lines. In the AP1000 design, while each PXS subsystem is associated with its own containment recirculation screen, the screens are cross connected. Likewise, each PXS subsystem is associated with its own IRWST screen, which is cross connected to a third IRWST screen. Therefore, if one PXS subsystem does not draw water both containment recirculation screens or all three IRWST screens, will be available to support the functioning subsystem. The NRC recommended providing two sumps to establish clear compliance with GDC 35, which requires long-term mitigation capability of a loss of coolant accident assuming a single failure. However, because the screens are either protected by pipe whip restraints (containment recirculation screens) or not identified as essential targets for the dynamic effects of pipe breaks (IRWST screens), and because the screens and their anchorage are designed to withstand seismic loads and postaccident operating loads, including head loss and debris weight, the staff finds there is no credible chance of failure of the screens, and the design meets the requirements of GDC 35.

NUREG-1793, Section 3.9.3, includes an evaluation of the structural adequacy of the sump screens using guidance from RG 1.82.

6.2.1.8.2.12 Upstream Effects

Any potential effects that debris may have in transit from its source to either the IRWST screens or containment recirculation screens are termed “upstream effects.” An evaluation of upstream effects ensures that flow necessary for recirculation is not held up by debris blockage at drains or other narrow pathways.

The applicant discussed upstream effects in its September 22, 2009, response to RAI-SRP6.2.2-SPCV-23. For ADS-4 discharge and break locations in the loop compartment, the limiting flowpath is the 2.3 m (7.5 ft) wide corridor between loop compartments, which is large enough to preclude debris blockage. For break locations at the maintenance floor elevation, three open stairwells will preferentially drain the break discharge to the sump. Curbs around openings of the two PXS rooms and the CVS room prevent water from entering these rooms. There are no LOCA break locations inside the CVS room, but such locations are present in the PXS room. If a break occurs in the PXS room, the room will fill and overflow onto the maintenance floor elevation, where the water will drain to the sump through the open stairwells. Breaks in the pressurizer line will either flow to the refueling cavity or to the IRWST, while a break in the passive residual heat removal heat exchanger tube will flow to the IRWST. Initially, water in the refueling cavity will drain to the containment sump, but this gravity-driven drain will cease when the water level in the sump increases enough to close the check valves on this line. An increase to the level of water in the IRWST could overflow into the refueling cavity, which will then drain to the containment sump as previously described.

For the AP1000, the potentially significant choke points for flow holdup are the gravity-driven drain lines and check valves in flowpaths between containment compartments. These drain lines and check valves are those in the refueling cavity drain lines, the PXS-A drain line, the PXS-B drain line, and the CVS compartment drain line. If recirculation water flow is restricted by any of these lines or valves, excessive amounts of water may be held up in the compartments and cavities, and the floodup level in the containment, which affects the gravity-driven core cooling flow, could be adversely affected. The staff asked about the issue in RAI-SRP6.2.2-SPCV-27 and RAI-SRP6.2.2-CIB1-31, and the applicant responded in letters dated January 29, 2010, and June 30, 2010. In these responses, the applicant stated that absent a LOCA in the PXS-A, PXS-B, and CVS rooms, there is no forward flow to transport debris to the lines that would prevent closure of the check valves. The staff verified that these valves are periodically stroke tested to ensure that the valves are capable of adequately reseating. The staff finds this acceptable because no debris would exist in these drain lines. Other sources of leakage into the rooms, such as through cracks in the wall, were determined to be insignificant. If an LOCA occurs in one of the PXS rooms (there are no break locations in the CVS room), the room would flood and debris could enter the drain lines. However, in this event the valves do not have to function, because the long-term cooling analysis includes cases that model a lower flood level to specifically account for water holdup resulting from a break in the PXS room. Even though the check valves in the drain lines are not expected to leak any significant amount, the applicant's wall-to-wall flooding analysis assumes that each line leaks 26.5 liters per minute (Lpm) (7 gpm). The staff finds that this is a conservative assessment.

For the refueling cavity drain lines, the check valves may be required to reclose after initially flowing forward following an LOCA and the water in the cavity may contain some debris. The applicant stated that the debris would not excessively erode or plug the valve, because the debris is a limited amount of latent and MRI debris and there is time for significant amounts to settle out. The applicant also stated that the debris could include some MRI fine particles that could pass through the downward-turned elbow drain line inlet and that this debris would not likely collect in the check valves. The staff finds that this is acceptable, because very little of the debris in the refueling cavity would pass into the drain lines and valves and would not significantly impede necessary drainage.

To address the issue of reclosure of these check valves after a period of initially flowing forward with debris in the lines, the applicant performed an analysis demonstrating that adequate long-term cooling was available even with a relatively large back leakage through this line. The analysis included an evaluation of Sensitivity Cases 3 and 10 from APP-PXS-GLR-001, which demonstrated that the margin in the original analysis, coupled with the long time required for the containment sump to leak through the check valve into the reactor cavity, was more than sufficient to remove the decay heat. The staff finds that the assumed leakage for this analysis is acceptable because the valves will be periodically stroke-tested to ensure that they acceptably reseat before debris conditions occur, and because the extent to which each closed check valve would have to be held open for this leakage to occur is significantly greater than what would be expected for these debris conditions. The staff reviewed the supporting calculation notes during its June 1 audit of APP-PXS-M3C-012, Revision 1, "Post-LOCA Refueling Canal Drain Check Valve Leak Evaluation," dated May 2010. The staff found the assumptions used to calculate flood levels, decay heat, and reduced core flow to be appropriate and conservative, and the staff agrees that the analysis demonstrates adequate core cooling flow with the assumed leakage.

The only potential choke point for flow to the IRWST is at the IRWST gutter, which extends around the entire containment circumference and drains to the IRWST through two 10-cm (4 in) pipes. Although there is a rough screen on top of the gutter to prevent large debris from entering, even if both discharge pipes become clogged, the only holdup volume will be water inside the gutter. This is because any excess water will spill over to the containment sump and recirculate through the containment recirculation screens. The applicant stated in its May 13, 2009, response to RAI-SRP6.2.2-SPCV-18 that the gutter volume is 0.96 m^3 (34 ft^3), corresponding to an IRWST water level change of 0.5 cm (0.2 in), which the staff finds is insignificant with respect to the recirculated volume.

In conclusion, the staff finds the applicant's evaluation of upstream effects acceptable.

6.2.1.8.2.13 Ex-Vessel Downstream Effects

In Section 6.2 of the AP1000 DCD, the applicant described the potential effects of sump debris on containment and core cooling, including the ex-vessel downstream effects. The term "downstream effects" refers to the effects of debris that passes through the recirculation screens on systems, structures and components located downstream of the recirculation screens. These effects have been evaluated for operating plants using data and methods developed by the PWR Owners Group. For the AP1000 plant, the applicant performed both an ex-vessel and an in-vessel evaluation for the AP1000 downstream effects. The ex-vessel evaluation describes the effects of debris on the system and components outside the core. This evaluation looks specifically at the disruption of the long-term core cooling flowpath (outside the core) by debris. Section 6.2.1.8.2.9 of this report addresses the staff's evaluation of in-vessel downstream effects (inside the core) for the AP1000.

The staff reviewed several documents, including information provided in the DCD and Westinghouse topical reports, TRs, and other letter reports related to the AP1000 design. Additionally, in an SER dated December 20, 2007, for ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," issued 1983, the NRC evaluated the methodology in WCAP-16406-P, Revision 1, for addressing downstream effects (both in-vessel and ex-vessel) of debris on the performance of containment and core cooling systems in existing U.S. PWRs. TR-26 addresses the applicability of WCAP-16406-P, Revision 1, to potential ex-vessel downstream effects for the AP1000 design.

As stated in TR-26, the data and methods used by the applicant to evaluate ex-vessel downstream effects are outlined in Revision 1 of WCAP-16406-P. The applicant stated that the evaluation methods identified in WCAP-16406-P, Revision 1, that are applicable to ex-vessel long-term core cooling recirculation flowpaths associated with the AP1000 design include valve evaluations for plugging and erosive wear, as described in Sections 7 and 8 and Appendix F of WCAP-16406-P, Revision 1. The screening criteria for valves that are identified in Revision 1 to WCAP-16406-P are applicable to valves in the long-term core cooling recirculation flowpath of PWRs. The applicant stated that only the explosively actuated (squib) valves in the post-LOCA flowpath are not covered by the screening criteria, but that once the squib valves are open, they very closely exhibit the characteristics of a standard gate valve.

The applicant stated that the AP1000 has design features that eliminate the need for downstream effects evaluations of components that are included in Revision 1 of WCAP-16406-P. Those evaluations excluded in the applicant's evaluation of the AP1000 design and the bases for their exclusion are as follows:

- Pump evaluations, including hydraulic performance, disaster bushing performance, and vibration analysis. Because of its passive design features, there are no safety-related pumps in the AP1000 passive core cooling flowpaths to evaluate.

- Heat exchanger evaluations for both plugging and erosive wear. There are no safety-related heat exchangers in the AP1000 passive core cooling flowpaths.

- Orifice evaluations for plugging and erosive wear as described in Sections 7 and 8 and Appendix F of WCAP-16406, Revision 1. There are no orifices in the post-LOCA recirculation flowpath of the AP1000 design.

- Settling of debris in instrumentation lines as described in Section 8 of WCAP-16406, Revision 1. There are no instrumentation lines used in the AP1000 post-LOCA containment recirculation flowpath design that are required to support a safety-related function; they are therefore excluded from consideration.

- Containment spray system. The AP1000 does not have a conventional containment spray system. The nonsafety containment spray function is not permitted to be used during a DBA and is therefore excluded from consideration for the AP1000 design.

As documented in TR-26, use of the applicable methods and models in WCAP-16406-P, Revision 1, consistent with the applicable amendments, limits, and conditions of the associated NRC safety evaluation on WCAP-16406-P, demonstrates that the AP1000 PXS equipment used in post-LOCA recirculation is acceptable for the expected debris loading in the recirculating fluid resulting from a postulated LOCA.

The staff finds that the applicant's evaluation of the ex-vessel downstream effects for the AP1000 design addressed the piping and valves in the recirculation path of the PXS. The methodology and acceptance criteria used are described in WCAP-16406-P, Revision 1, and are consistent with the applicable amendments, limits, and conditions of the NRC safety evaluation for WCAP-16406-P, Revision 1, and are, therefore, acceptable to the staff.

The applicant identified equipment in the post-LOCA flowpath using current piping and instrumentation diagrams (P&IDs) for the AP1000 PXS. The AP1000 PXS P&IDs show no pumps, heat exchangers, orifices, and spray nozzles in the PXS. Therefore, although included in the method of WCAP-16406-P, Revision 1, the applicant’s evaluation of the AP1000 PXS does not address pumps, heat exchangers, orifices, spray nozzles, and instrumentation tubing because these components and features are not included in the design of the AP1000 PXS. The applicant listed and described components that are in the AP1000 long-term core cooling flowpath, including both the containment recirculation flowpath and the IRWST injection flowpath.

The downstream path evaluation includes the effects of erosion and wear of the component materials and plugging of the various components. In order to apply erosive and abrasive wear rate models, the debris size and concentration were first assessed. The applicant evaluated the debris types and stated that the debris was composed of latent fibrous and particulate material, with a small amount of coatings debris. In RAI-SRP6.2.2-CIB1-07, the staff raised questions regarding: (1) the results of the applicant’s evaluation of the piping system for plugging and wear; (2) the composition of possible debris other than the evaluated latent and coatings debris; and (3) the effects the composition may have on the downstream flowpath. In a response dated November 11, 2008, the applicant clarified that there is no other debris that would be transported to the recirculation screens. The staff finds that the applicant’s assumed composition of latent and coating debris for the ex-vessel downstream evaluation is consistent with the debris generated from coatings and the chemical effects evaluated by the staff for the AP1000.

Each identified valve in the PXS was evaluated for plugging and wear against the applicable initial screening criteria in WCAP-16406-P, Revision 1. The PXS consists of open gate, check, and squib valves, all of which are greater than 2.54 cm (1 in) based on their individual flow line diameters. Therefore, according to the initial screening criteria, the valves do not need further evaluation for plugging or wear. However, the criterion to be greater than 2.54 cm (1 in) is based on the assumption that only debris of a certain size will pass through the containment recirculation screen because of the size of its holes. For a limiting direct-vessel-injection-line-break LOCA in the AP1000 design, the water level in the containment permits direct entry of debris-laden water into the DVI line at the break location. This could result in a significantly higher concentration of debris and larger pieces of debris entering into the core cooling flowpath than if all cooling water first passed through the screen. Therefore, in RAI-SRP6.2.2-CIB1-08, the staff asked about the assumed concentration and composition of debris for the DVI-line-break LOCA.

In a response dated November 6, 2008, the applicant stated that the only ex-vessel components that would see unfiltered containment water during a DVI line break are the 17.3 cm (6.8 in) ID DVI pipe and the reactor vessel DVI nozzle that has a 10.16 cm (4-in) ID venturi. The applicant stated that these components have openings that are large enough that plugging will not occur and that wear of the venturi is not an issue because the purpose of the venturi is only to limit high-pressure blowdown flow, not to restrict recirculation. The staff finds that the applicant has provided reasonable assurance that plugging and wear of components because of debris-laden water in the ex-vessel PXS flowpaths are not significant concerns for the AP1000 design.

All instrumentation sensors in the PXS recirculation lines are strapped to the outside of the piping. Therefore, there are no instrumentation tubes or sensing lines to evaluate for potential debris collection in the tubes or sensing lines. In RAI-SRP6.2.2-CIB1-09, the staff asked about the possible effects of debris, chemicals, and gases in the recirculation water on the accuracy of these strapped instruments as a result of changing the velocity of sound in the fluid. In a response dated November 6, 2008, the applicant clarified that the only strapped-on instruments are temperature sensors that are not affected by the velocity of sound in the fluid. The staff finds that the applicant's clarification adequately addressed the staff's question.

The applicant also evaluated the potential debris collection in the PXS flowpath piping. Based on the minimum flow rates for the PXS flow lines, the applicant determined that the transverse velocity is sufficient to prevent debris settlement in the PXS flow lines; therefore, blockage in PXS flow lines from the settling out of debris would be precluded. In RAI-SRP6.2.2-CIB1-10, the staff asked about flow rates that could be less than the minimum value assumed (e.g., during system flow initiation or realignment) and whether significant debris settlement could occur that would prevent necessary system core cooling flow. In a response dated November 6, 2008, the applicant responded that with the small amount and low concentration of debris that would be present following a LOCA and the large-diameter piping, any settling out of debris, even during very small flow, such as during startup or realignment, would have a negligible effect on PXS flow resistance. The staff finds this acceptable and agrees that because of the very low concentration of debris in the water, only a very small amount of debris could settle out during low flow conditions and would not cause significant blockage.

The applicant credited only passive systems for core and containment cooling; however, the AP1000 design includes a nonsafety active system (the RNS) that could be used for removing core and containment heat during various plant conditions, including a LOCA. The staff evaluated the possible effects of operation of these nonsafety active system ex-vessel downstream components and their capability to remove heat for long-term core cooling. As a result, in RAI-SRP6.2.2-CIB1-01, the staff asked about the use of these systems and: (1) the effects of possible additional amounts of debris ingested as a result of use of the active systems; (2) how ingested debris could affect the capability of these active systems when relied on for long-term cooling; (3) how ingested debris could affect these active systems for long-term cooling; (4) how ingested debris could affect the pressure integrity, leakage, and containment isolation function of these active systems; and (5) whether leakage through pump seals or other components could increase local dose rates so that credited operator actions, if any, would not be met. In a response dated November 11, 2008, the applicant stated that the evaluation of downstream effects has already included the conservative assumption that all latent containment debris could be ingested.

The applicant also stated that it evaluated the RNS ex-vessel downstream components, including pump seals, for wear, abrasion, and erosion using the evaluation methods of WCAP-16406-P for the assumed debris and found the components to perform their functions for a 30-day period beginning at the time of recirculation from the sump. The applicant further stated that in the event of a large source term (like the design-basis core melt source term), the RNS is automatically (i.e., not manually) isolated from the containment. Additionally, the applicant submitted a response dated May 13, 2009, to the staff's followup RAI-SRP6.2.2-CIB1-23 regarding the capability of the RNS isolation valves to close and not leak excessively under debris-laden conditions after the RNS has been functioning. The applicant performed an evaluation of the effects of wear, abrasion, debris loading, and erosion and concluded that the isolation valves will close and not leak excessively under these conditions. The staff finds that this adequately addresses the concerns regarding the evaluation of ex-vessel downstream RNS components under debris-laden conditions.

In RAI-SRP6.2.2-CIB1-11 and RAI-SRP6.2.2-CIB1-12, the staff asked about possible blockage of the ex-vessel downstream flowpath into the vessel and out of the vessel back to the break location as a result of settling or precipitation of boric acid and other chemicals. In responses dated November 6 and 11, 2009, the applicant stated that the concentration of boron and other chemicals is low enough that their precipitation would not occur over the 30-day mission time. The staff finds that this adequately addresses the issue of potential blockage of ex-vessel core cooling water flowpaths by boron or chemical precipitation.

The applicant performed an evaluation of the effects of the possible collection of noncondensable gases in high points in the PXS flowpath. Gases in sufficient quantities that collect and are trapped at high points could cause unacceptable pressure losses and restriction of system cooling flow, especially in a gravity-driven system. In RAI-SRP6.2.2-CIB1-13, the staff asked about the possible collection of noncondensable gases in the PXS flowpath that could impede cooling flow. In a response dated November 11, 2008, the applicant stated that the CMT and passive residual heat removal heat exchanger circuits are not susceptible to gas accumulation during preaccident standby conditions, except in an engineered high point pipe stub that has redundant level sensors. A low concentration of hydrogen is dissolved in the RCS, which is separated from these circuits by a single valve, but very little pressure is required to maintain the hydrogen in solution. The accumulator circuits are also expected to contain nitrogen dissolved in the water, but there is substantial dP following a LOCA such that any gas pockets would not impede accumulator flow. Once the accumulator water is emptied, nitrogen from the tank will be injected into the RCS. This nitrogen is readily vented from the RCS through the open ADS valves, and integral testing performed for the AP1000 showed that this nitrogen injection did not adversely affect the plant and system performance. The recirculation flowpaths from the IRWST to the RCS or from the recirculation screens to the RCS either contain water all the time or, if they contain air, they are short, straight horizontal pipes, such that the air is expelled and will readily fill with water. The staff finds this acceptable for addressing the possible effects of noncondensable gases, other tha
n those gases that could form as a result of chemical reactions or gases that come out of solution at higher accident-condition temperatures, which are discussed below.

Noncondensable gases that could form in the ex-vessel recirculation flowpath as a result of chemical reactions and additional gases that may come out of solution at higher accident temperatures were also evaluated. In RAI-SRP6.2.2-CIB1-23, the staff asked about the possible effects such gases could have in restricting cooling flow. In a response dated November 11, 2008, the applicant stated that the amount of gas as a result of chemical reactions is small and is limited by the small amount of materials that could react with the coolant. The amount of gases that could form by chemical reactions and that could come out of solution at higher temperatures would have time to bubble to the surface in the pool and be released to the containment. The staff finds this acceptable, because the amount of these gases is limited in quantity or they will have time to come out of solution in the pool water before recirculating in the core cooling flowpath.

Another issue of concern to the staff is how the squib valves may differ from the evaluated gate valves and the effects that squib valve propellant residue or chemicals could have on the ex-vessel downstream flowpath. The staff notes that the actuation of the squib valves occurs when there is no debris in the valve with which these chemicals could interact, such that a combined effect of both chemicals together with debris is not possible. However, the effects of the residue or chemicals as they mix with the system fluid without debris could also be an issue. Therefore, in RAI-SRP6.2.2-CIB1-25, the staff asked about the differences from the gate valve design and the possible effects the residue or chemicals could have in impeding the recirculation flow through the valves. In a response dated September 22, 2009, the applicant stated that both the squib valves and gates valves have sufficiently large flow openings and internal crevices, such that latent and particle debris would not get caught or restrict the flow. Regarding the effects of the residue and chemicals, the applicant stated that each of the 12 squib valves in the plant contains less than approximately 300 g (0.66 lbs) of propellant in which potassium is the predominate constituent. If all of the resulting potassium were mixed with the minimum post-LOCA sump volume, the concentration of potassium would be approximately 0.5 ppm. The applicant stated that this is much less than the concentration of sodium that would be present, which is chemically similar to potassium. The applicant stated that at this concentration, the potassium would stay in solution, and there would be negligible impact on downstream components. The applicant also stated that the remaining constituents of the propellant include gases in concentrations much lower than the potassium concentration, such that they would have no impact on the downstream components.

In reviewing Tier 1 to the AP1000 DCD, the NRC staff found that Table 2.1.2-4, “Inspections, Tests, Analyses, and Acceptance Criteria,” included ITAAC to verify the design and qualification of the as-installed squib valves in the ADS of the AP1000 reactor. However, AP1000 DCD Tier 1, Table 2.2.3-4, “Inspections, Tests, Analyses, and Acceptance Criteria,” did not include ITAAC to verify the design and qualification of the as-built squib valves in the AP1000 PXS. In RAI-SRP6.2.2-CIB1-29, the NRC staff noted that the squib valve design and applications in the AP1000 reactor will not be adequately similar to be represented by the tests or type tests for the ADS squib valves. Therefore, the NRC staff requested that the applicant provide additional ITAAC to verify the capability of the different squib valve designs and sizes and their applications in the AP1000 reactor. In its RAI response dated March 12, 2010, the applicant agreed that ITAAC related to the active safety-related valve functions of the PXS containment recirculation squib valves and IRWST injection squib valves should have been included in AP1000 DCD Tier 1, Table 2.2.3-4. The applicant provided a planned revision to AP1000 DCD Tier 1, Section 2.2.3, “Passive Core Cooling System,” to address the PXS squib valves. The applicant also provided planned ITAAC for the PXS squib valves to be included in AP1000 DCD Tier 1, Table 2.2.3-4, specifying that tests or type tests will be performed that demonstrate the capability of the valve to operate under its design condition, and that an inspection will be performed for the existence of a report verifying that the as-installed squib valves are bounded by the tests or type tests. Revision 18 to the AP1000 DCD included the modifications described in the RAI response. The NRC staff finds that AP1000 DCD Tier 1, Section 2.2.3, and Table 2.2.3-4, provide acceptable ITAAC to verify the design and qualification of the as-built PXS squib valves. Therefore, RAI-SRP6.2.2-CIB1-29 is resolved.

The applicant's evaluation of the effects of the residue and chemicals on the sump pool water is acceptable to the staff, since the concentrations are very small. However, the residue and chemicals from the valve actuation initially would be introduced into a much smaller volume of water inside the valve and in the immediate vicinity of the downstream pipe. To resolve this issue, the staff is reviewing the specific qualification testing of the squib valves being performed by the applicant. The staff has observed actual testing of prototype squib valve designs and will also observe testing of the final squib valve designs before their installation in the plant. Based on the staff's observations, the amount of residue and chemicals entering the flowstream area is small and would not likely restrict the cooling flow through the squib valves. Therefore, the staff finds the applicant's evaluation and the squib valve qualification testing program to adequately address the possible effects of squib valve residue and chemicals on the ex-vessel downstream flowpath.

In conclusion, the staff finds the applicant's evaluation of ex-vessel downstream effects acceptable.

6.2.1.8.3 Conclusion

In summary, the staff reviewed the proposed DCD changes and associated COL requirements to establish the adequacy of the IRWST and containment recirculation sump screen performance. Based on the evaluation described in the foregoing sections, the staff concludes that the design and analyses satisfy the requirements in GDC 35, GDC 38 and 10 CFR 50.46(b)(5).

6.2.2 Passive Containment Cooling System

6.2.2.1 Summary of Technical Information

Revision 17 of the DCD makes several changes to the text and figures in Section 6.2.2, and Change Number 70, included in a letter dated July 6, 2010, proposes further changes to the figures. As a result of these changes, the applicant also revised DCD Tier 1, Figure 2.2.2-1, "Passive Containment Cooling System Piping and Instrumentation Diagram (PCS P&ID)," and Table 2.2.2-2, as described and evaluated below.

6.2.2.2 Evaluation

Changes made to Figure 6.2.2-1 are evaluated as follows:

- Two passive containment cooling water storage tank (PCCWST) discharge lines were combined to a single larger line. TR-103, "Fluid System Changes," issued May 2006, stated that this change was necessary in order to achieve the flow rates required for adequate containment cooling and was not a functional change. In a July 18, 2008, response to RAI-SRP6.2.2-SPCV-01, the applicant confirmed that the flow rates reported in Table 6.2.2-1 were unaffected; therefore, the staff finds this piping change to be acceptable.

- The two PCCWST narrow-range pressure-based level sensors that shared taps with the wide-range level sensors were replaced with two inside wall-mounted ultrasonic, noncontact level sensors. TR-103 reported that this change was made in order to enhance accuracy over the wide-range level measurement at the top of the tank. Based on DCD Tier 1, Table 2.2.2-1, the narrow-range sensors are not safety related; therefore, the change does not impact the safety design basis for the PCS. Additionally, the change to ultrasonic sensors provides diversity in measurement of the PCCWST level because the nonsafety-related ultrasonic sensors now differ from the safety-related dP sensors. The staff finds this change acceptable.

- A second makeup line to the PCS water distribution bucket was added to provide the piping separation required to support a beyond-DBA scenario. Because a safety-related line already exists to supply the water distribution bucket, this new line is nonsafety related. The staff finds this change acceptable because it has no impact on the safety design basis for the PCS.

- As described in TR-103, a nonsafety-related spray system was added to the spent fuel pool (SFP) in response to a National Academy of Sciences study on the potential danger if water were drained from the SFP. The line to the new spray system is branched from the existing PCS makeup line to the SFP, and the location where this existing line stems from the PCCWST is changed from a 15.24 cm (6 in) standpipe to the bottom of the tank. A normally closed manual isolation valve was added to provide a boundary between the ASME Code, Section III Code Class 3 SFP makeup line and the nonsafety-related SFP spray header. Both this valve and the existing normally closed safety-related manual isolation valve must be opened in order to activate the SFP spray. In Revision 0 of its response to RAI-SRP6.2.2-SPCV-02, dated July 18, 2008, the applicant stated that the use of the SFP spray was controlled by procedure and not allowed during a DBA; therefore, there would be no impact on PCS performance. The staff requested additional information to determine if PCCWST inventory could be distributed to the SFP when it was needed for containment cooling. In Revision 1 of its response to RAI-SRP6.2.1.1-SPCV-06, dated August 31, 2009, the applicant added a paragraph to DCD Section 6.2.2.4 stating that the use of the PCCWST to provide water to the SPF spray header would be governed by the Extensive Damage Mitigation Guidelines included in NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," issued December 2006, and this document was added as Reference 33 to DCD Section 6.2.7. The staff finds this acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- As described in a letter dated July 6, 2010, Change Number 70 revises the P&ID to accommodate lower than expected efficiency of the PCS recirculation pumps. These changes, which include increasing the size of the recirculation pump piping and associated valves, and adding a line to bypass the pumps, ensure that the ancillary diesel generator does not exceed the steady-state load limit. These changes are acceptable because they have no impact on the system or component design evaluated in NUREG-1793, Section 6.2.2.

- Several changes were made to the P&ID that the staff agrees have no impact on the evaluations made in NUREG-1793, Section 6.2.2. These include moving the recirculation heater from downstream to upstream of the chemical addition tank, standardizing flow orifice flange sets to 1.9 -cm (0.75–in) piping components, and correcting errors in the certified P&ID.

The changes made in the Revision 17 of the DCD to Tier 2, Figure 6.2.2-2, "Simplified Sketch of Passive Containment Cooling System," and Tier 1, Figure 2.2.2-1, are consistent with the items described above. The staff finds them acceptable.

DCD Tier 1, Table 2.2.2-1, identifies the components that perform safety-related functions for the PCS, and Table 2.2.2-2 identifies the PCS safety-related lines. Revision 17 of the DCD added several valves to Table 2.2.2-1 to reflect the P&ID changes previously discussed, but the lines in Table 2.2.2-2 were unaltered. However, the applicant subsequently proposed changes to the lines in this table in its July 18, 2008, and May 27, 2009, responses to RAI-SRP6.2.2-SPCV-04 and RAI-SRP6.2.2-SPCV-20. The changes made to both Tier 1 tables were found to be acceptable because they are consistent with changes made to Tier 2, Figure 6.2.2-1, and with each other. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In the certified DCD, Section 6.2.2.4 states that the frequency of operational testing of the PCS was consistent with the plant TS and the inservice testing program. Revision 17 of the DCD removes the reference to the plant TS. However, the applicant's May 27, 2009, response to RAI-SRP6.2.2-SPCV-03, Revision 1, withdraws this change. The staff finds this acceptable because the TS are used to define some of the operational test frequencies. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

6.2.2.3 Conclusion

The staff finds the proposed changes to the DCD acceptable because they have no impact on the statements or conclusions of NUREG-1793, Sections 6.2.1.6 and 6.2.2, regarding primary containment testing and inspection and the PCS, respectively.

6.2.3 Shield Building Functional Design

6.2.3.1 Summary of Technical Information

Modifications were made to the shield building to strengthen it against additional external hazards, to make it more robust to seismic events, and to simplify construction. These changes, which include alterations to the inlet and outlet of the air flowpath and the height of the shield building, are expected to impact the air flow rate through the PCS. APP-GW-GLR-096, submitted August 10, 2010 includes a description of these changes and the impact they have on existing test reports. It also documents how the changes were incorporated into a WGOTHIC model that was used to run the limiting DBA and beyond design basis accidents (BDBA) affected by these changes. Appendix A of APP-GW-GLR-096 includes proposed DCD changes.

6.2.3.2 Evaluation

The staff evaluation of APP-GW-GLR-096 will be documented in Chapter 23.

6.2.3.3 Conclusions

The impact of the shield building changes on the containment functional design capability and the staff's conclusion will be provided in Chapter 23.

6.2.4 Containment Isolation System

The major function of the containment isolation system (CIS) is to isolate the containment to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary. The CIS consists of the piping, valves, and actuators that isolate the containment.

In a letter dated April 5, 2006, the applicant submitted a request to modify the AP1000 DCD Tier 2 information to incorporate the ability to inject a small quantity of zinc acetate into the RCS. TR-32 describes this proposed change.

Zinc acetate would be added using the same piping and valving as the CVS hydrogen addition piping, which contains two containment isolation valves, CVS-PL-V092 and CVS-PL-V094. This would also require changing the normal position of the outside containment valve, CVS-PL-V092, from normally closed to normally open. The inside containment isolation valve, CVS-PL-V094, is a normally closed check valve. The proposed hardware change would also replace a portion of the 2.54 cm (1 in) pipe (downstream from the inside containment isolation valve) with a heavier wall 1.27 cm (0.5 in) pipe. The staff has reviewed and approved this change (Section 5.2.3 of this report).

6.2.4.1 Summary of Technical Information

The modification described in TR-32 results in changing the outside containment isolation valve, CVS-PL-V092, from normally closed, failed closed to normally open, failed closed.

Revision 17 of the DCD provides four additional overpressure relief valves between two normally closed containment isolation valves, identified in Table 6.2.3-1, “Containment Mechanical Penetrations and Isolation Valves.” These valves have also been added to Tier 1, Figure 2.2.1-1, and identified in Tier 1, Table 2.2.1-1, as CCS-PL-V220, SFS-PL-V067, VWS-PL-V080, and WLS-PL-V058.

6.2.4.2 Evaluation

The normal position of the containment isolation valve, CVS-PL-V092, in the CIS would change from normally closed to normally open. The valve would still fail closed, maintaining its containment isolation function. This change is indicated in Revision 17 of the DCD, Tier 2, Table 6.2.3-1, and shown in Figure 9.3.6-1, Sheet 1 of 2. The following functions and properties are not affected: the containment isolation function signal, the containment isolation design, the valve designation as active (Table 3.9-12 in the DCD), the safety-related mission, the inservice testing type and frequency requirements (Table 3.9-16), and the valve functional requirements for containment isolation (Tier 1, Table 2.3.2-1).

GDC 55, “Reactor Coolant Pressure Boundary Penetrating Containment”; GDC 56, “Primary Containment Isolation”; and GDC 57, “Closed Systems Isolation Valves,” of Appendix A to 10 CFR Part 50 require that containment isolation valves outside containment be located as close to containment as practical. Acceptance Criterion 9 in NUREG-0800 Section 6.2.4, “Containment Isolation System,” Revision 3, issued March 2007, also invokes this requirement. In RAI-SRP6.2.4-SPCV-01, the staff asked the applicant to provide the distances from the containment to the outboard isolation valves, and in RAI-SRP-6.2.4-SPCV-02, the staff asked the applicant to add the approved distances to DCD Tier 2, Table 6.2.3-1. The applicant responded to the RAIs with the distances provided in its letter dated January 13, 2009, and with its commitment in its letter of April 13, 2009, to add these distances to DCD Tier 2, Table 6.2.3-1. The staff found the responses acceptable. The corresponding ITAAC in Table 14.3-7, “Radiological Analysis,” that the containment penetration isolation features be configured as given in Table 6.2.3-1, remains acceptable to the staff as written. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In Revision 15 of the DCD, additional requirement M in Section 6.2.3.1.3, “Containment Isolation Design,” states the following:

> Containment penetrations with leak-tight barriers, both inboard and outboard, are designed to limit pressure excursions between the barriers due to heating of fluid between the barriers. The penetration will either be fitted with relief or check valves to relieve internal pressure or one of the valves has been designed or oriented to limit pressures to an acceptable value.

Section 6.2.4.2 of NUREG-1793 states, “All overpressure relief valves used as containment isolation valves comply with the SRP acceptance criterion of having a setpoint greater than or equal to 150 percent of the containment design pressure.”

The applicant's response to RAI-SRP6.2.4-SPCV-03 confirms that the four new relief valves comply with the NUREG-0800 Section 6.2.4 acceptance criterion of having a setpoint greater than or equal to 610 kPa (88.5 psig), 150 percent of the containment design pressure. These new relief valves are shown in Revision 17 of the DCD, Tier 1, Figure 2.2.1-1, and listed in Tier 2, Table 6.2.3-1.

In RAI-SRP6.2.4-SPCV-03, the staff also asked whether the CVS letdown line at penetration P06 should be similarly provided with an overpressure relief valve between the two normally closed containment isolation valves, CVS-PL-V045 and CVS-PL-V047. The applicant’s response of May 20, 2009, indicates that it will add relief valve CVS-PL-V058, which will comply with the design requirements of the relief valves already added. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Among the apparent discrepancies noted in RAI-MISC-SPCV-01 is the fact that the change to the normal position of containment isolation valve CVS-PL-V092 has not been reflected in Section 9.3.6, “Hydrogen Addition Containment Isolation Valve,” which should indicate that the valve is normally open, failing closed. Also, the position of CVS-PL-V092 in Table 9.3.1-1, “Safety-Related Air Operated Valves,” should read “normally opened, failed closed.”

In Revision 16 of the DCD, the applicant also made two editorial changes to Table 6.2.3-1, Sheet 1 of 4. The first identifies containment isolation valves CAS-PL-V015, CCS-PL-V201, CCS-PL-V208, and CCS-PL-V207 and observes the correct naming convention. In addition, for containment isolation valves CCS-PL-V207 and CCS-PL-V208, the applicant corrected the containment isolation signal to “S.” The staff agrees with these two editorial changes.

6.2.4.3 Conclusion

The staff finds that the applicant’s proposed modification to the AP1000 CVS system, approved in Section 9.3.6, does not adversely affect the containment isolation design and is, therefore, acceptable. The five thermal relief valves provided for overpressure protection are in accordance with regulatory guidance, consistent with DCD commitments, and are acceptable to the staff.

The editorial changes are acceptable.

6.2.5 Containment Hydrogen Control System

The containment hydrogen control system is provided to limit the hydrogen concentration in the containment so that containment integrity is not endangered.

On September 2004, the staff provided its assessment of the AP1000 hydrogen ignition subsystem design in Section 6.2.5.1 of NUREG-1793. As stated in paragraphs 3 and 9 of Section 6.2.5.1, adequate igniter coverage was provided based on implementation of the igniter location criteria in DCD Table 6.2.4-6.

DCD Tier 2, Table 6.2.4-6, provides the criteria used in the evaluation and the application of the criteria to specific compartments. On the basis of the staff's review and the applicant's implementation of the igniter location criteria as listed in DCD Tier 2, Table 6.2.4-6, the staff concluded that adequate igniter coverage had been provided.

6.2.5.1 Summary of Technical Information

In APP-GW-GLN-003 (TR-37), "Hydrogen Igniter Locations," Revision 1, the applicant modified the elevations or locations of certain hydrogen igniters within the AP1000 hydrogen control system. The applicant stated that the modifications were necessary because either the polar crane elevation or the pressurizer height had been changed, or in order to place the igniters in more easily accessible locations or to avoid trip hazards.

In Revision 16 of the DCD, Figures 6.2.4-5 through 6.2.4-13 show the proposed locations of the hydrogen igniters, and Tables 6.2.4-6 and 6.2.4-7 identify the proposed hydrogen igniter locations. The number of igniters is unchanged at 64.

6.2.5.2 Evaluation

Revision 16 of the DCD, Table 6.2.4-6, provides the criteria used in the evaluation and the application of the criteria to specific compartments. The changes to igniter locations as a result of the continuing COL and detailed design activities for the AP1000 satisfy the igniter location criteria identified in DCD Table 6.2.4-6 (Sheet 1 of 3) that were used for the DC review of the hydrogen igniter subsystem and referenced in the AP1000 SER. Therefore, changes in the placement of the hydrogen igniters that are consistent with the criteria in Table 6.2.4-6 do not alter the design function of the igniters, have no effect on any analysis or analysis method, and do not affect the performance or controls of hydrogen control functions.

On the basis of the staff's review and the applicant's implementation of the igniter location criteria as listed in DCD Tier 2, Table 6.2.4-6, the staff concludes that adequate igniter coverage has been provided.

6.2.5.3 Conclusion

The staff finds that the applicant's proposed modification to the AP1000 hydrogen control system design with respect to the change in hydrogen igniter locations, as described in TR-37, is consistent with the previously approved criteria and, therefore, acceptable.

6.2.6 Containment Leak Rate Test System

The containment leak rate test system is designed to verify that leakage from the containment remains within limits established in the TS.

6.2.6.1 Summary of Technical Information

The containment penetrations, including electrical penetrations, subject to Type B testing appear in Figure 6.2.5-1, "Containment Leak Rate Test System Piping and Instrumentation Diagram." The applicant has added the test connection assembly for the newly added electrical penetration, P03, to the list of electrical penetrations test connections in Figure 6.2.5-1.

6.2.6.2 Evaluation

The design commitment to provide a test assembly for Type B leak rate testing for the newly added electrical penetration, P03, is acceptable.

6.2.6.3 Conclusion

Based on its review, the staff finds the proposed addition of a Type B leak rate test assembly for the new electrical penetration, P03, acceptable.

6.2.8 Tier 1, Chapter 2.2.1, Containment System

6.2.8.1 Summary of Technical Information

In TR-97, APP-GW-GLN-022, Revision 1, "DAS Platform Technology and Remote Indication Change" dated May 2007, the applicant identifies and justifies standard changes to Revision 15 of the DCD. These changes include relocating the diverse actuation system (DAS) squib valve control cabinet (DAS-J3-003) and adding the DAS instrumentation cabinet (DAS-JD-004) to the southern section of the auxiliary building. The DAS is a nonsafety-related system. These changes necessitate the addition of a containment electrical penetration, P03. In a letter dated May 14, 2007, the applicant submitted responses to all the NRC RAIs on TR-97.

6.2.8.2 Evaluation

The staff's assessment of the CIS design was provided in Section 6.2.4 of NUREG-1793. As stated in the NUREG-1793 section, the containment penetration design of isolation barriers met the following acceptance criteria of NUREG-0800 Section 6.2.4.

Containment isolation equipment may be subject to potentially harsh conditions resulting from pressure, temperature, flooding, jet impingement, radiation, missile impact, and seismic response. The staff's review confirmed that the CIS had been properly classified to ensure that protection from these environmental hazards is encompassed by the mechanical and electrical design bases and quality standards of the isolation system.

The CIS will be designed to ASME Code Section III, Class 2 criteria. Containment penetrations are classified as Quality Group B, as defined in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4 and seismic Category I. The containment penetrations are identified as Class B, equivalent to ANS safety Class 2. The staff concluded that the applicant had selected the appropriate mechanical design classification for the CIS.

The staff determined that the CIS met the acceptance criteria of Section 6.2.4 of NUREG-0800, including the relevant requirements of GDC 1, "Quality Standards and Records"; GDC 4; and GDC 16, "Containment Design."

The acceptable design standards for electrical penetrations, namely ASME Code Section III, seismic Category I, non-Class 1E qualified, harsh environment qualified, will also apply to the new electrical penetration, P03, as shown in DCD, Revision 17, Tier 1, Table 2.2.1-1. The addition of the new electrical penetration, P03, is shown in DCD Revision 17, Figure 2.2.1-1 where it is included in the note "1 of 24 and one spare." This is inconsistent with TR-97 and in RAI-SRP6.2.4-SPCV-04, the staff requested the note be corrected. In a letter dated May 17, 2010, the applicant agreed to revise the note to state "1 of 25." In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Tier 1, Table 2.2.3-4, Inspections, Tests, Analyses, and Acceptance Criteria, describes the equipment listed in Tier 1, Table 2.2.3-6, including all electrical penetrations, as having sufficient thermal lag to withstand the effects of hydrogen burns associated with severe accidents. The newly added electrical penetration, P03, has been added to the list of electrical penetrations in Tier 1, Table 2.2.3-6. The design commitment to provide the thermal lag to the newly added electrical penetration, P03, is acceptable.

6.2.8.3 Conclusion

Based on its review, the staff finds the proposed change as described in TR-97, Revision 1, which adds an additional containment electrical penetration in accordance with previously acceptable design criteria for electrical penetrations, namely ASME Code Section III, seismic Category I, non-Class 1E qualified, harsh environment qualified, will also apply to the new electrical penetration, P03. The staff finds that the applicant's proposed modification to the AP1000 CIS design with respect to the addition of an electrical penetration, as described in TR-97, is consistent with the previously approved criteria and is, therefore, acceptable.

6.4 Control Room Habitability Systems

6.4.1 Summary of Technical Information

Section 6.4 of the AP1000 DCD has undergone significant revision. The revisions include a major redesign of the AP1000 main control room (MCR) emergency habitability system (VES). The VES is a passive system design that consists of safety-related canisters of air that supply the control room with fresh, uncontaminated breathing air. The system does not require alternating current (ac) power to function and is required to function for 72 hours. After 72 hours, nonsafety systems can be credited for control room habitability. In the certified design, the COL applicant was responsible for the testing frequency associated with the control room integrity program. Additionally, the certified design provided no filters to remove radioactivity from the control room environment. The system only replaced the contaminated control room air with bottled air that was uncontaminated. In the amendment, the applicant removed the COL information item and provided a general description of the control room integrity program with the testing frequency. Additionally, to permit the use of the AP1000 design at more sites, the radiation dispersion factors were also expanded. In developing a control room integrity program, which includes in-leakage testing for the control room envelope (CRE), the applicant was unable to establish achievable acceptance criteria for the in-leakage testing with the new radiation dispersion factors. As a result, the applicant made a series of significant design changes to add margin to the control room dose calculations.

The changes included reducing control room in-leakage through various design provisions, including precluding ductwork from penetrating the CRE and reconfiguring the vestibule. A filter train was added to the passive system as well. The filter train consists of an eductor with ductwork, silencers, a particulate filter, and a high-efficiency particulate air (HEPA) filter. The design change resulted in changes to Tier 1, Tier 2, TS, and ITAAC. Additionally, the applicant has made a number of other unrelated changes, including the redesignation of the technical support area to the control support area. This allows COL applicants more flexibility in designating a technical support area. The design of this system has evolved over the course of the review, and numerous applicant submittals describe the changes. The applicant consolidated all the changes into a response to RAI-SRP6.4-SPCV-15 Revision 1, which was submitted in a letter dated May 24, 2010.

The changes can be grouped into six broad categories. The first includes changes associated with addressing the control room integrity program in the DCD. The approved version of the DCD made this the responsibility of the COL applicant and documented it as a COL information item. The second involves changes associated with the new passive filter train. Third are changes associated with the design changes to reduce the unfiltered in-leakage. The second and third categories of changes were necessary because the applicant revised the radiation dispersion factors to expand the scope of sites that would be covered by the certification. The higher dispersion factors required less leakage and more effective control room fission product removal. The fourth category includes changes associated with the redesignation of the technical support center (TSC) as the control support area. The fifth involves changes intended to improve operational flexibility of the system by including four isolable banks of compressed air canister banks rather than a single bank. Last, there are editorial changes.

A number of changes are associated with the removal of the COL information item on the control room integrity program. The applicant provided a description of the control room integrity program in the final safety analysis report (FSAR) using tracer gas testing. The applicant also included TS implementing the program.

A number of changes are associated with the introduction of a passive filter train. An innovative single passive filter train was added to the existing compressed air system. An eductor was designed to connect to the existing compressed air system. The eductor draws in unfiltered control room air and circulates it through a filter train. The filters are designed and tested to meet the intent of RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Revision 3, issued June 2001, and design provisions were made to demonstrate that the single passive train meets the single failure criteria.

A number of design changes are intended to reduce unfiltered in-leakage. The control room air exhaust was moved to vent through the vestibule. This created a sort of purge, reducing unfiltered air ingress from the MCR doors. Additionally, the applicant made design commitments to preclude ductwork from penetrating the CRE. Other material changes were made as well.

Changes are associated with the redesignation of the TSC as the control support area. The applicant changed the description in DCD Tier 2, Section 6.4, and the Tier 2 description in Chapter 18.

The applicant made changes to improve operational flexibility. For example, it divided the existing air tanks into four isolable headers to allow maintenance work to be done on some of the tanks while the remainder of the system remains functional. TS were added to address the situation where one bank of compressed air tanks is out of service. The applicant also performed the dose analysis for the fuel handling accident with a shorter time to reactor shutdown. This permits fuel movement earlier than was previously analyzed.

The applicant also made a number of editorial changes. For example, it changed the general information for onsite chemicals identified in Table 6.4-1. The list of chemicals is general and does not provide the quantity of material or the distance to the control room intake. Additionally, the COL applicant is responsible for identifying and evaluating the onsite and offsite chemicals. As a result, the staff considers this change to be editorial. This SER does not describe these types of changes in detail. With the exception of the editorial changes, a more detailed evaluation of the major changes is provided below.

6.4.2 Evaluation

6.4.2.1 Evaluation of Control Room In-Leakage Testing

In developing a control room integrity program, the applicant had difficulty establishing an achievable in-leakage that could be demonstrated through testing. Originally, the applicant proposed a total effective in-leakage of 0.0425 cubic meters per minute (m^3/min) (1.5 cubic feet per minute (cfm)). This would account for both unfiltered in-leakage as well as effective in-leakage though the doors. The staff issued a number of RAIs to determine how this design limit would be demonstrated. In response, the applicant made a major redesign of the VES. In a letter dated May 24, 2010, the applicant responded to RAI-SRP6.4-SPCV-15 R1 and included all the DCD changes associated with the VES redesign. The applicant has removed the analysis assumption of 1.5 cfm effective unfiltered in-leakage from the MCR dose analysis. The analysis assumption was revised to assume 5 cfm unfiltered in-leakage into the control room as a result of ingress/egress activities. An unfiltered in-leakage of 5 cfm is appropriate for the AP1000 control room because of the incorporation of a two-door vestibule. The control room doses with this increased effective in-leakage assumption required the addition of a passive filtration line to the VES to remain below regulatory limits.

The NRC issued GL 2003-1, "Control Room Habitability," dated June 12, 2003, to alert addressees to findings that the control room licensing and design bases and applicable regulatory requirements may not be met, and that existing specification surveillance requirements may not be adequate. In 2006, the staff approved a modification to the Standard Technical Specifications (STS) (NUREG-1430, "Standard Technical Specifications-Babcock and Wilcox Plants"; NUREG-1431, "Standard Technical Specifications-Westinghouse Plants"; NUREG-1432, "Standard Technical Specifications-Combustion Engineering Plants";

NUREG-1433, "Standard Technical Specifications-General Electric Plants (BWR/4)"; and NUREG–1434, "Standard Technical Specifications-General Electric Plants (BWR/6)") that were proposed by the PWR and boiling-water reactor (BWR) owners groups' Technical Specification Task Force (TSTF) in STS change traveler TSTF-448, Revision 3. The notice of availability for adopting TSTF-448, Revision 3, using the consolidated line item improvement process was published in the *Federal Register* on January 17, 2007 (72 FR 2022). TSTF-448, Revision 3, addresses the tracer gas surveillance, adding a TS action for an inoperable CRE and instituting a CRE habitability program that will ensure that CRE habitability is maintained.

In Revision 17 of DCD Tier 2, Sections 6.4.5.1 and 6.4.5.4 commit to performing tracer gas testing during preoperational inspection and testing, and periodically during the life of the unit in accordance with RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," issued May 2003; they also commit to performing followup self-assessments. However, the AP1000 generic TS did not include a surveillance requirement to measure unfiltered in-leakage into the CRE (the tracer gas test), and required actions for an inoperable CRE boundary and a CRE habitability program as approved by the NRC in STS generic change TSTF-448, Revision 3.

In RAI-SRP6.4-SPCV-01, the staff asked the applicant to incorporate the changes to the STS made by TSTF-448 in AP1000 DCD Tier 2, Chapter 16, "Technical Specifications." In a letter dated May 4, 2009, the applicant responded to incorporate the DCD changes according to the staff's request.

Additionally, the staff raised questions with the applicant about a demonstrable control room in-leakage design basis. VES is a passive system design. There is no safety-related emergency electric diesel to provide electrical power during DBAs accompanied by a loss of outside power. Therefore, there is not enough Class 1E power to drive fans to recirculate control room ventilation air through filters to remove activity during radiological accidents. There is a limited supply of bottled compressed air to maintain control room habitability during DBAs. Based on DCD Revision 17, the VES design had no filtration trains to recirculate and filter the air in the CRE. The bottled air can only supply 1.84 m^3/min (65 cfm) for 72 hours. To meet the dose limits, the unit could only accept 0.0425 m^3/min (1.5 cfm) MCR in-leakage (MCR unfiltered in-leakage and MCR doors ingress/egress combined) to maintain operator dose rates below the required levels. Safety class calculation "AP1000-LOCA Dose Analysis" is the key analysis to demonstrate that MCR operator dose is below 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE), as dictated by GDC 19, "Control Room," in Appendix A to 10 CFR Part 50. Should the MCR in-leakage exceed the 0.0425 m^3/min (1.5 cfm) design limit by a small amount, the 5 rem TEDE limit would be exceeded. The safety margin for this configuration was very small.

The applicant chose to ease the safety margin issue for the VES passive system design by installing a passive filtration line to the VES system. The bottled air will provide 1.84 m^3/min (65 cfm) to an eductor to induce at least 17.0 m^3/min (600 cfm) of air from the MCR into a filter bank to remove radionuclides. This will allow 0.425 (15 cfm) MCR in-leakage and keep MCR operator dose below the required 5 rem TEDE. By doing this, the safety margin issue of the VES passive system design is relaxed. The resulting system is a single safety Class 3, passive filtration line, which is new to the ASME Code. Using the passive filtration line, the VES is able to maintain operator dose below the GDC 19 requirements assuming a total in-leakage of 0.425 m^3/min (15 cfm). The dose analysis assumes that 5 cfm results from ingress/egress activities and up to 0.283 m^3/min (10 cfm) of in-leakage can occur from all other sources. The 0.283 m^3/min (10 cfm) in-leakage is verified through the control room integrity leakage program.

As a result, the applicant has established an achievable design basis in-leakage, and it has properly accounted for effective in-leakage through the doors. Additionally, there is a control room integrity program that meets the recommendations in RG 1.197, and there are TS consistent with TSTF-448. Therefore, the staff finds the applicant's approach acceptable.

6.4.2.2 Evaluation of the Passive Filter Train

The VES uses a bank of compressed air storage tanks to provide the MCR with breathable air and maintain a positive pressure relative to its adjacent areas during accident conditions. The system is designed to deliver a constant flow of 1.84 ± 0.14 standard m^3/min (65 ± 5 standard cubic feet per minute (scfm)) for 72 hours. Using the current VES design, the applicant developed a passive air filtration line that uses an eductor to induce a filtration flow through the MCR of at least 17.0 standard m^3/min (600 scfm). The components that comprise the passive filtration portion of the VES are located entirely within the MCR envelope.

To the extent applicable, the filtration line is designed in accordance with RG 1.52, Revision 3. The applicant added a conformance assessment to Appendix 1A to the DCD to compare the passive filtration line to the requirements defined in RG 1.52, Revision 3, and added this RG to DCD Table 1.9-1. In the conformance assessment, the applicant took an exception to Regulatory Position C.6.1. The staff asked the applicant to explain why it needed an exception to this regulatory position. In a letter dated May 24, 2010, the applicant provided a revision to its response to RAI-SRP6.4-SPCV-15 that stated that the filtration line conforms to Regulatory Position C.6.1. It included the associated changes to Appendix 1A to the DCD to reflect this conformance. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The filtration line comprises an intake grill located near the inner vestibule door inside the MCR envelope. This location was chosen because it is expected to be the location where the greatest amount of in-leakage occurs as a result of ingress/egress activities. Flow is then directed through safety-related ductwork and a silencer into the eductor. The VES supply line is connected to the eductor. The VES bottle air supply is the motive force that drives at least 600 cfm of air flow through the intake duct. The eductor works by directing a small amount of compressed air through a nozzle along the walls of the opening. When released into the nozzle, this small amount of compressed air is moving at near sonic speeds and creates a powerful vacuum in the area upstream of the nozzle. This vacuum draws air in through the surrounding duct and pulls it through the nozzle. The compressed air then carries it downstream away from the eductor. The eductor then directs the flow through a second silencer, a HEPA filter, a charcoal filter, and a postfilter. The filtration units work to remove particulates and iodine from the air to reduce the potential MCR dose. Redundant flow instruments are located downstream of the filtration units to ensure that adequate flow is passing through the filtration units during testing activities. The new flow instrumentation has high and low alarms to alert the operator of possible filtration issues during testing. After passing the flow instrumentation, the filtered air is discharged to three locations inside the MCR envelope. Approximately 1.70 to 1.98 m^3/min (60–70 cfm) would be discharged in the vicinity of the shift supervisor’s officer or operator break room, and the remaining 17.0 m^3/min (600 cfm) would be discharged into the main control area through two discharge paths on the opposite side of the control room from the air intake located in the operations work area. Two flow dampers located downstream of the postfilter control the flow distribution.

Although each of the components in the filter train arrangement has been used before, this particular application is novel in the nuclear industry. The applicant constructed a scale model to demonstrate that the system would function. The scale model was tested, and the test results were presented to the NRC in a public meeting on December 15, 2009. The tests demonstrated that the system would function as designed.

The existing pressure-regulating valves in the VES control pressure and flow from the emergency air storage tanks into the eductor. The pressure at the outlet of the valve is controlled via a two-stage, self-contained pressure control operator. A failure of either stage of the pressure-regulating valve will not cause the valve to fail completely open. A failure of the second stage of the pressure-regulating valve will increase flow from the emergency air storage tanks. There is adequate margin in the emergency air storage tanks such that an operator has time to isolate the line and manually actuate the alternative delivery line.

To reduce the overall noise, the silencers, eductor, and filtration unit must all be located behind the main control area. The applicant added an ITAAC to DCD Tier 1, Section 2.2.5, to verify that the noise levels in the MCR remain below the recommended guidelines in NUREG-0700, "Human-System Interface Design Review Guidelines," issued May 2002. The ITAAC will verify that the noise level at the operator work station will remain below 65 decibels when the VES is in operation.

A filtration flow of at least 17.0 m^3/min (600 cfm) resolves the original difficulty in having a control room design in-leakage that could be demonstrated through testing. At a filtration flow rate of 17.0 m^3/min (600 cfm), 0.425 m^3/min (15 cfm) is an acceptable MCR in-leakage to maintain operator dose rates below the required levels. (The total flow at the outlet of the filters must be at least 17.0 m^3/min (600 cfm) plus the flow from the compressed air tanks.) This allows the dose analysis to assume a constant 0.142 m^3/min (5-cfm) in-leakage through the vestibule and up to 0.283 m^3/min (10 cfm) of in-leakage from sources other than through the vestibule. Using a tracer gas test, it would be possible to verify that the in-leakage from sources other than the vestibule is less than 0.283 m^3/min (10 cfm). The applicant revised the existing ITAAC related to tracer gas testing (ITTAC 7b in Section 2.2.5-5 of the DCD) to reflect the appropriate acceptance criteria of less than or equal to 0.283 m^3/min (10 scfm). A technical specification has been added to the AP1000 technical specifications to incorporate the requirements of TSTF-448 for a CRE habitability program.

The staff raised a number of technical issues in its review of the passive filter system, as described below.

6.4.2.2.1 Evaluation of Issues Associated with the Eductor in the Passive Filtration Line

Nuclear power plant applications have limited operational and maintenance experience with the eductor. The frequency of the technical specifications surveillance test of the eductor should be based on experience with eductor system degradation. However, the frequency chosen for the surveillance was not supported by a technical rationale or data on the degradation of eductors. Therefore, in RAI-SRP6.4-SPCV-09, the staff asked the applicant to justify the surveillance frequency with a technical rationale that is based on data associated with eductor degradation.

The applicant responded in a letter dated December 9, 2009, stating that the "frequency of Technical Specification Surveillance Testing for the Main Control Room Emergency Habitability System (VES) eductor was chosen to align with the Ventilation Filter Testing Program (VFTP) identified in Surveillance Requirement 3.7.6.11 of the AP1000 Technical Specifications as revised by RAI-SRP-6.4-SPCV-06." Additionally, the applicant provided examples of operational data on eductors in industrial applications. This information supports the applicant's claims, and the staff finds the applicant's proposed TS surveillance testing frequency for the eductor reasonable.

6.4.2.2.2 Evaluation of Issues Associated with the HEPA Filter in the Passive Filtration Line

Section 6.3, regarding HEPA filter in-place leak testing, of RG 1.52 shows the acceptable combined penetration and leakage (or bypass) to be less than 0.05 percent of the challenge aerosol. The applicant's proposed TS 5.5.13 shows this value to be 0.5 percent. In its response to RAI-SRP6.4-SPCV-06 received in a letter dated May 4, 2009, the applicant stated that each HEPA filter cell is individually shop-tested to verify an efficiency of at least 99.97 percent in accordance with ASME AG-1, Section FC. The staff asked the applicant to provide a technical basis to credit 99.97 percent HEPA filter efficiency at 0.5 percent penetration and system bypass conditions.

The applicant responded in a letter date May 24, 2010 that the markup of the TS that indicates the combined penetration and leakage of less than 0.5 percent is an editorial error. The applicant corrected TS 5.5.13 in the draft DCD revision pages to indicate a leakage value of less than 0.05 percent.

In accordance with Section 6.3 of RG 1.52, to be credited with 99 percent removal efficiency for particulate matter in accident dose evaluation, a HEPA filter bank should demonstrate an aerosol leak test result of less than 0.05 percent of the challenge aerosol. It its response to RAI-SRP6.4-SPCV-06, the applicant stated that the HEPA filters will remove 99 percent of particulates consistent with the guidance in RG 1.52. The staff asked the applicant to provide a technical basis to credit 99 percent HEPA filter efficiency for particulate matter in accident dose evaluation at 0.5 percent penetration and system bypass conditions.

In a letter dated February 25, 2010, the applicant responded to RAI-SRP6.4-SPCV-10, Revision 2, by stating that it intends to comply with Section 6.3 of RG 1.52, Revision 3, as indicated in the markup of Appendix 1A in RAI-SRP6.4-SPCV-06, Revision 0. The staff found the DCD changes provided in the response to RAI-SRP6.4-SPCV-15, Revision 1 to be in alignment with the guidance and acceptable.

Section 6 of RG 1.52, Section 9.5 of ASME N510-2007, "Testing of Nuclear Air Treatment Systems," and Section 5.7 of ASME N511-2007, "In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems," specify dP testing across the HEPA filter bank. However, proposed TS 5.5.13 did not specify the dP testing across the HEPA filter bank. The staff asked the applicant to explain why the technical specifications did not specify the rationale for the dP testing across the HEPA filter bank.

The applicant responded in a letter dated May 24, 2010 in RAI-SRP6.4-SPCV-15, Revision 1 that it would include dP testing across the combined HEPA filter, charcoal adsorber, and the postfilter in the ventilation filter testing program. The staff verified that the draft DCD revisions included this. As a result, the staff finds the applicant's approach acceptable.

6.4.2.2.3 Evaluation of Issues Related to the Adsorber in the Passive Filtration Line

Section 6.4, regarding adsorber in-place leak testing, of RG 1.52, Revision 3, shows the acceptable combined penetration and leakage (or bypass) to be less than 0.05 percent of the challenge gas. The applicant's response to RAI-SRP6.4-SPCV-15, provided in a letter dated February 25, 2010, proposed TS 5.5.13, which shows this value to be 0.5 percent. In its response to RAI-SRP6.4-SPCV-06, the applicant stated that the charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME AG-1 and RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 2, issued June 2001. Both RG 1.52 and RG 1.140 specify a combined penetration and leakage (or bypass) adsorber in-place leak test criterion of 0.05 percent or less of the challenge gas. The staff asked the applicant to provide the technical basis for the exception taken to relax the adsorber penetration and system bypass criterion from 0.05 percent to 0.5 percent.

The applicant responded to RAI-SRP6.4-SPCV-11, Revision 1 in a letter dated December 3, 2009, by stating that it intends to comply with Section 6.3 of RG 1.52, Revision 3, as indicated in the markup of Appendix 1A in RAI-SRP6.4-SPCV-06. The applicant noted an editorial error in the markup of the technical specifications that indicates combined penetration and leakage of less than 0.5 percent. The applicant corrected TS 5.5.13 in Reference 13 to indicate a leakage value of less than 0.05 percent. As a result, the staff finds the approach acceptable.

For maximum assigned credit for active carbon decontamination efficiencies of 95 percent (elemental iodine and organic iodide), RG 1.52, Revision 3, Section 7, regarding laboratory testing of charcoal samples, shows an acceptable penetration of less than 2.5 percent for a 5.08 cm (2-in)-deep charcoal bed. For maximum assigned credit for active carbon decontamination efficiencies of 99 percent (elemental iodine and organic iodide), Section 7 shows an acceptable penetration of less than 0.5 percent for a 10.16 cm (4-in) bed. In its response to RAI-SRP-6.4-SPCV-06, the applicant stated that the charcoal filters would remove 90 percent of the elemental iodine and 30 percent of the organic iodine, claiming to be consistent with RG 1.52, Revision 2, issued March 1978. For the assigned activated carbon decontamination efficiencies of 90 percent (elemental iodine) and 30 percent (organic iodide), RG 1.52, Revision 2, Section 6, related to laboratory testing criteria for activated carbon, shows an acceptable laboratory testing criterion for a methyl iodide penetration of less than 10 percent for a 5.08 cm (2-in)-deep charcoal bed. The applicant's proposed TS 5.5.13 shows a value of 35 percent. The 35-percent allowable penetration should be calculated from a safety factor of 2 recommended by GL 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," dated June 3, 1999, (100 percent – organic iodide efficiency)/safety factor = (100 percent – 30 percent)/2 = 35 percent). The staff asked the applicant to provide the technical basis for assigning a credit for active carbon decontamination efficiencies of 90 percent (elemental iodine) and 30 percent (organic iodide) charcoal carbon efficiency at 35 percent penetration conditions.

The applicant responded to RAI-SRP6.4-SPCV-11, Revision 1 by stating that the "technical basis for assigning activated carbon decontamination efficiencies of 90 percent (elemental iodine) and 30 percent (organic iodine) for charcoal carbon efficiency at 35 percent penetration conditions should have been identified as Reference 1." Reference 1 identifies a methodology described in GL 99-02 to calculate the allowable penetration percentage based on assumed organic iodine efficiency and a defined safety factor. Using the provided methodology, a 35 percent penetration condition is calculated assuming a 30 percent organic iodine efficiency and a safety factor of 2. The staff noted that the values in the table were generated for specific penetration, elemental and organic efficiencies, and residence times. The use of the 35 percent penetration value was not specifically approved by the staff in the RG and further justification would be needed to apply the methodology.

The applicant stated that it would revise TS 5.5.13 to show an in-place test of the charcoal adsorber with a penetration and system bypass less than 0.05 percent (RAI-SRP6.4-SPCV-10, Revision 2). Additionally, the applicant included the residence time as a design parameter in the DCD. RG 1.52, Revision 3, specifies the value of less than 0.05 percent. The applicant demonstrated that the value chosen for penetration relative to the efficiencies used in the dose analysis is conservative. The staff verified that the applicant included the change in the response to RAI-SRP6.4-SPCV-15, Revision 1 in the draft DCD revisions and finds the approach acceptable.

Section 6 of RG 1.52, Section 10.5 of ASME N510-2007, and Section 5.8 of ASME N511-2007 specify dP testing across adsorber banks. However, proposed TS 5.5.13 did not specify the dP testing across adsorber banks. The staff asked the applicant to provide the rationale for not specifying the dP testing across the charcoal filter bank in the TS.

The applicant responded to RAI-SRP6.4-SPCV-15, Revision 1 by stating that it would include dP testing across the combined HEPA filter, charcoal adsorber, and postfilter in the ventilation filter testing program. The staff verified that this was included in the draft DCD revisions. As a result, the staff finds the applicant’s approach acceptable.

The staff noted in Section 6.4.2.3 of NUREG-1793 that the applicant referenced RG 1.140 for the design, construction, and qualification of the charcoal adsorber. The staff notes that it cited RG 1.52, Revision 3, for all other aspects of the design. The staff is unclear why the applicant would use RG 1.140 for this specific aspect of the application rather than RG 1.52, Revision 3. In a letter dated May 24, 2010, the applicant revised its response to RAI-SRP6.4-SPCV-15 to state that RG 1.52, Revision 3, is the correct reference and provided an associated DCD change. The staff found the response acceptable.

6.4.2.2.4 Evaluation of the Test Frequency of the Combined Filters Pressure Drop

TS 5.5.13, which the applicant submitted in response to RAI-SRP6.4-SPCV-15, Revision 0, states that a "program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with regulatory guide 1.52, Revision 3, ASME N510-1989, and AG-1." Item d of TS 5.5.13 requires a test of "the pressure drop across the combined HEPA filter, the charcoal adsorber, and the post filter." However, RG 1.52, Revision 3, ASME N510-1989, and AG-1 do not have a clearly defined combined pressure drop test frequency. This dP measure is a surveillance test made at regular intervals to detect deterioration that may develop under service conditions. Regular in-place testing is necessary because deterioration may take place even when the system is not being operated.

For the combined HEPA filter, charcoal adsorber, and postfilter pressure drop test in TS 5.5.13.d, the staff asked the applicant to provide a specific citation and reference for performing this test and to provide the required test frequency. The applicant responded in a letter dated June 2, 2010, to RAI-SRP6.4-SPCV-16 by stating that it would revise the proposed TS 5.5.13 to list the frequencies for each test listed in TS 5.5.13.a, b, c, and d. It removed the reference to RG 1.52, Revision 3, and ASME N510-1989 from TS 5.5.13.d that were proposed in the response to RAI-SRP6.4-SPCV-15. The frequencies provided for Section 5.5.13.a, b, and c are the same as the frequencies listed in RG 1.52, Revision 2. As a result, the staff finds these frequencies acceptable. The test in TS 5.5.13.d will be conducted every 24 months, which aligns with the frequencies for the HEPA filter and charcoal adsorber inplace tests and the charcoal adsorber sampling and analysis (TS 5.5.13.a, b, and c). A specific RG or standard citation is not needed with the revised TS 5.5.13.d, since the test is described by the TS itself and the frequency is specified.

Because the frequencies given in TS 5.5.13.a, b, and c are those specified in RG 1.52 and the frequency given in TS 5.5.13 d is aligned with the frequencies in the RG, the staff finds the DCD changes identified in the response to RAI-SRP6.4-SPCV-16 acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

6.4.2.2.5 Evaluation of the Safety Class of Passive Filtration Flow Instrumentation

Redundant flow instruments are located downstream of the filtration units to ensure that adequate flow is passing through the filtration units during testing activities. The instrumentation is not safety-related. The applicant's rationale is that the instrument does not perform a safety function. An existing VES safety flow instrument indicates whether there is sufficient flow coming from the compressed air tanks to induce the passive filtration flow. The new flow instrumentation has high and low alarms to alert the operator of possible filtration issues during testing.

Section 3.3.1.3 of ANSI/ANS-51.1-1983 (Reference 2) states the following:

> safety class 3 (SC-3) shall apply to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the following nuclear safety functions:
>
>> k. Ensure nuclear safety functions provided by SC-1,-2, or -3 equipment, m. Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-1, -2, or -3 equipment"

At a filtration flow rate of 17.0 m^3/min (600 cfm), 0.425 m^3/min (15 cfm) is an acceptable MCR in-leakage level to maintain operator dose rates below the required levels. (The total flow at the outlet of the filters must be at least 17.0 m^3/min (600 cfm) plus the flow from the compressed air tanks.) This allows the dose analysis to assume a constant 0.142 m^3/min (5-cfm) in-leakage through the vestibule and up to 0.283 m^3/min (10 cfm) of in-leakage from sources other than through the vestibule. The 17.0 m^3/min (600 cfm) flow induced by the eductor is the design basis used by the dose calculation to make sure filtration units in the passive air filtration line will work to remove particulate and iodine from the air to reduce the potential MCR dose. This dose calculation is required to satisfy GDC 19 in Appendix A to 10 CFR Part 50.

The two flow instruments in the filtration line provide information to ensure the capability of the eductor to draw at least 17.0 m^3/min (600 cfm) so the VES system safety function (MCR habitability during radiological accidents) can be achieved. The existing VES safety flow instrumentation to indicate whether there is sufficient flow (1.84 m^3/min (65 cfm)) coming from the compressed air tanks to induce the passive filtration is not a direct indication of the performance of the eductor.

The operators would rely on this instrumentation during an accident to ensure that the safety-related filtration train was functioning. Based on ANSI/ANS-51.1-1983, the staff asked that at least one flow instrument in the passive air filtration line be safety related. The staff also asked the applicant to provide additional justification that the operators would not rely on this instrumentation during an accident or else make one of the instruments safety related.

The applicant responded in a letter dated February 25, 2010, to RAI-SRP6.4-SPCV-15, Revision 0, by stating that the operator would not rely on this instrumentation during an accident. Rather, the operator would rely on the flow instrumentation directly from the compressed air tanks. The staff noted that this instrumentation would not notify the operator of a clogged or partially blocked component in the filter train and that flow blockage was a credible failure after 24 hours. The applicant responded by making design changes demonstrating compliance with the single failure criteria. Further discussion of the single failure criteria appears below. Given that the applicant eliminated all credible single failures, the staff agrees that reliance on the flow instrumentation from the compressed air tanks during an accident is acceptable. As a result, the two redundant flow instruments that are located downstream of the filtration units to ensure that adequate flow is passing through the filtration units during testing activities do not need to be safety-related.

6.4.2.2.6 Evaluation of the Single Failure of the Passive Filtration Line

The applicant stated that redundant passive filtration lines are not required because the passive filtration line has no electrical power requirements, contains no moving parts, and requires no maintenance such as adjusting setpoints or lubricating bearings. The applicant stated that adequate design margin is provided to prevent the likelihood of a passive failure.

Section 3.2.1.c of ANSI/ANS-51.1-1983 states that fluid systems required to support, directly or indirectly, the three nuclear safety functions stated above shall be capable of performing these functions as provided in ANSI/ANS-58.9-1981, “American National Standard Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems.”

ANSI/ANS-58.9-1981 defines “passive failure” as “a failure of a component to maintain its structural integrity or the blockage of a process flow path.” In this standard, the term refers to a random failure, its consequential effects assumed in addition to an initiating event, and its consequential effects for the purpose of safety-related fluid system design and analysis. This standard defines rules for application of the single failure criteria. During the short term, the single failure considered may be limited to an active failure. During the long term, assuming no prior failure during the short term, the limiting single failure considered can be either active or passive. “Long term” is defined as that period of safety-related fluid system operation following the short term, during which the safety function of the system is required. “Short term” is defined as that period of operation up to 24 hours following an initiating event.

Additionally, TSTF-448 and the associated TS Bases describe the expectation to consider passive failures. Specifically the TSTF states, “No single active or passive failure will cause the loss of outside or recirculated air from the CRE [control room envelope].”

The VES passive filtration line is required to meet single failure criteria. The single active or passive failure of a component in the VES passive filtration line, assuming a loss of outside power, shall not impair the ability of the system to perform its design function. The staff noted that if the present passive filtration design proposed by the applicant has a passive failure (e.g., the nozzle section of the eductor fails to induce the minimum 17.0 m^3/min (600 cfm) flow), the safety function of the filtration may not be achievable. The proposed VES passive filtration line does not have independent, redundant trains to recirculate and filter the CRE. The staff asked the applicant to provide a justification that the described system meets the single failure criteria or to provide a redundant filter train. The applicant responded to RAI-SRP6.4-SPCV-13, Revision 0 in a letter dated December 11, 2009 by stating the following:

> The primary components that comprise the main control room habitability system (VES) passive filtration line are duct work, two silencers, an eductor, and a filtration unit. The passive filtration line has no active components. The only active components in the VES are in the air delivery portion of the system that provides the motive flow to induce to the filtration flow. The air delivery portion of the system also provides breathable air for main control room occupants during abnormal scenarios. In the air delivery portion of the system, there are redundant flow paths. The redundant flow paths prevent a single active or passive failure from impairing the ability of the system to perform its design function. Based on the guidance in SECY-77-439, the passive filtration portion of the system must be evaluated for a credible passive failure 24 hours after the start of an event. SECY-77-439 defines a passive failure as events such as a line blockage or structural failure of a static component that limits the effectiveness of the component. Though a passive failure in the passive filtration portion of the VES is highly unlikely, it would not impair main control room habitability. Dose analysis for the AP1000 main control room was performed to verify that in the event of a passive failure in the passive filtration portion of the VES 24 hours after the initiation of the event operator doses would remain below 5 rem TEDE. The limiting AP1000 main control room dose scenarios were evaluated for a loss of filtration flow 24 hours into an accident. These scenarios are limiting since they involve a release 24 hours after the initiation of the event. The analysis showed the following acceptable increases in dose rates compared to the scenarios when filtration is available for 72 hours. Therefore, the passive filtration portion of the VES can sustain a single passive failure without impairing main control room habitability for the first 72 hours following a design basis accident.

The applicant demonstrated that filtration was not needed in the long term or after 24 hours. The staff noted that this response adequately addressed the blockage of the filter or ductwork. However, it did not address the potential blockage of the eductor, which would prevent the required breathable air from entering the control room. These concerns were brought to the attention of the applicant. In a letter dated February 25, 2010, the applicant submitted a response to RAI-SRP6.4-SPCV-13, Revision 2, which incorporate a bypass line for the eductor and thereby provides operators with the ability to deliver the air flow from the emergency air storage tanks following the highly unlikely passive single failure of the eductor. The design addition added a flow control orifice and normally closed manual valve to allow the 65 scfm of breathable air to flow into the MCR without passing through the eductor. Also, the applicant added a normally open manual valve upstream from the eductor to provide full isolation of the flowpath to the educator. The revised dose analysis demonstrates that air filtration is not necessary in the long term. The manual bypass valves with the safety-related flow meter from the compressed air tanks allow the air delivery function to be accomplished if there is a blockage. As a result, the staff finds the applicant’s response acceptable because the design is capable of withstanding the unlikely credible single failures in the passive filter train.

The applicant has designed a filter system that meets, to the extent practicable, the guidance of RG 1.52. The applicant proposed surveillance and testing consistent with the intent of RG 1.52 and the STS. The design was demonstrated to function in scale-model testing. The applicant has addressed all credible single failures. As a result, the staff finds the single filter train acceptable.

6.4.2.3 Evaluation of Design Changes To Reduce Unfiltered In-Leakage

The applicant changed the MCR envelope purge design. The VES discharge air flow is now directed into the entry vestibule to provide a continuous vestibule purge. This helps to reduce the radioactivity introduced into the MCR each time there is access to or from the MCR during a radiological accident. The VES discharge dampers originally discharged through the MCR wall directly to the atmosphere outside the MCR. This design change redirects the damper discharge flow into the MCR vestibule and adds openings to allow free passage from the vestibule to the hallway.

The applicant made a number of other changes to the design of the CRE. For example, the applicant eliminated ductwork penetrating the CRE. The applicant also installed isolation capability for various control room penetrations, like the sanitary system and normal vents. Additionally, the applicant proposed different materials for the control room. Each of these changes improves the overall design, and the staff finds them acceptable. The staff notes that the overall effectiveness of the CRE is demonstrated through the testing associated with the control room integrity program.

During the course of the review, the applicant proposed a number of other changes. For example, the applicant proposed an additional actuation to isolate the normal control room ventilation system. The applicant also proposed to remove the dose calculations associated with the normal ventilation system. According to TR-122, “AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-122 (TR-122),” Revision 0, dated July 27, 2007; the DCD did not report the dose analysis for the nuclear island non-radioactive ventilation system (VBS) operating cases because the VBS does not continue to operate if a high-2 radioactivity level is detected. The staff was concerned that the normally operating active ventilation system (VBS) in the supplemental air filtration mode should not be isolated early and should be demonstrated to meet the regulatory limits. With active systems, the VBS is the preferable system to function because it will provide greater comfort to the operators and will provide active cooling for the control room equipment. In response to RAI-TR122-SPCV-01, the applicant stated that it will update the DCD to present the results for the case with the VBS operating for the duration of the accident and remove the earlier isolation of the VBS.

The applicant also proposed to procedurally delay access into the control room. According to TR-122, Revision 0, the AP1000 requires procedurally delaying access to the control room in order to meet the operator dose limits. The staff was concerned that GDC 19 requires that adequate radiation protection shall “be provided to permit access and occupancy of the control room under accident conditions.” The AP1000 control room, as proposed by TR-122, did not seem to provide adequate radiation protection to permit access under accident conditions because of the access control process. In its response to RAI-TR122-SPCV-01, submitted in a letter dated December 17, 2008, the applicant stated that it would update the DCD to present the results for the case without consideration for an MCR entry time delay. The results will show that the AP1000 design with the vestibule purge configuration fully satisfies GDC 19, with or without credit for the use of the entry time delay. The staff finds this response acceptable.

6.4.2.4 Redesignation of Technical Support Center

The applicant proposed to redesignate what was referred to as the TSC in the certified design to the control support area. The change removed the TSC from the Section 6.4 description in Tier 2. If a reactor is being built on a site that already has an operating reactor and a functioning TSC, it is reasonable for the COL applicant to want a single TSC. The applicant added a statement to Tier 2* that requires the TSC to be located in the control support area. As a result, if a COL applicant would like to locate the TSC somewhere other than in the control support area, a Tier 2* departure requiring NRC approval would be necessary. In a subsequent letter dated January 27, 2010, the applicant removed the Tier 2* information and included a statement in DCD Chapter 18 indicating that the location of the TSC would be in the control support area. Section 13.3 of this report approves this change. Because the information is Tier 2, rather than Tier 2*, a departure may not require prior NRC approval. However, the location of the TSC is part of the Emergency Plan, which a COL application is required to include under 10 CFR 52.79(a)(21), "Contents of applications; technical information in final safety analysis report." As a result, regardless of whether the Tier 2 departure would require NRC approval, the Emergency Plan and the location of the TSC therein would be subject to prior NRC review and approval. The analysis showing that the control support area remains habitable following an accident, if the nonsafety-related VBS system is available, remains in the DCD. The analyses were redone in the amendment with the higher dispersion factors, and the results remain within regulatory limits. Although these values were removed from the DCD in one of the intervening revisions, the DCD now refers to these analyses as the "VBS Operating" dose results. Because there are no impacts from a safety perspective associated with this change, the staff finds the change acceptable.

6.4.2.5 Changes to Improve Operational Flexibility

TS 3.9.7 specifies that the minimum radioactive decay time ensures that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are consistent with the assumptions in AP1000 DCD Tier 2, Chapter 15.

In TR-122, the applicant changed the minimum decay time mandated in the TS to 48 hours. This TS change affects the convenience of plant operations; there are no system design concerns. The staff finds this change acceptable based on an acceptable finding associated with the related dose calculations that are being reviewed by the Office of New Reactors, Division of Site and Environmental Reviews, Technical Specification Branch.

Supplementary changes have also been made to VES to increase the reliability of the system and provide a safe environment for workers performing maintenance on the system. Because there was concern about isolating this system for repair or maintenance with a single valve, double isolation valves have been added to the banks of air tanks and the supply lines. Individual fill and vent lines have also been added to each bank of tanks to allow one bank to be taken out of service and be recharged while keeping the other three banks of tanks online and ready for service at all times. Revised Figure 6.4-2 in the DCD depicts these changes.

The applicant modified the condition in TS 3.7.6 for "VES inoperable" to specify a condition for "One bank of VES air tanks (8 tanks) inoperable." The completion time to increase pressure in the operable tanks to the upper portion of the system operating band is 12 hours. The completion time is 7 days to restore the VES to operable status.

In the revised TS bases (B 3.7.6), the applicant stated the following:

> If one bank of VES air tanks (8 tanks out of 32 total) is inoperable, then VES is able to supply air to the MCR for 54 hours (75 percent of the required 72 hours). If VES is actuated, operator must take actions to maintain habitability of the MCR once the air in the tanks has been exhausted. The VBS supplemental filtration mode or MCR ancillary fans are both capable of maintaining the habitability of the MCR after 54 hours.
>
> Increasing the pressure in the OPERABLE tanks from the minimum pressure of 3400 psig to the upper portion of the system operating band maximizes the time the VES will be able to supply air to the MCR. The 12-hour Completion Time provides sufficient time to achieve the increased pressure.
>
> With one bank of VES air tanks inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE VES air tanks, along with compensatory operator actions, are adequate to protect the main control room envelope habitability. The 7 day Completion Time is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the reactor core, the low probability radioactivity release, and that the remaining components and compensatory systems can provide the required capability.

The staff reviewed the rationale. At 75 percent VES air capacity (54 hours); the system no longer accomplishes the safety function for 72 hours. Additionally, the VBS supplemental filtration is not safety-related. The applicant's regulatory treatment of non-safety systems (RTNSS) (ancillary fans) may not be available in 54 hours. Additionally, increasing the pressure in the operable tanks is not a reviewed operating condition and may not be advisable with a degraded system. Further, restoring or replacing an inoperable tank is a relatively simple evolution, and it is not clear why 7 days is needed to complete this action.

The 72-hour design basis for passive safety system capability and operator actions 72 hours after accident initiation has been evaluated as part of the RTNSS process. The safety-related design basis for the VES is to operate for 72 hours. The 54-hour VES air capacity is in a condition outside the accident analysis. If one bank of VES air tanks (8 tanks out of 32 total) is inoperable, the loss of safety function would merit more immediate action.

The staff asked the applicant to explain why such a long completion time is appropriate for the loss of a safety function to restore operability.

The applicant responded in a letter dated December 3, 2009, to RAI-SRP6.4-SPCV-14, Revision 0 by stating the following:

> Any leakage that would cause the system to enter the Technical Specification of "One bank of VES air tanks (eight tanks) inoperable" would most likely be at the safety-relief valves (V040A/B/C/D). The air bank with the excessively leaking relief valve will be individually isolated and depressurized to allow the safety-relief valve to be replaced. Because only the air bank with the leaking safety-relief valve is isolated, the remaining three air banks will be at or above the minimum operating pressure of the system. The maintenance on an individual air bank can be done without affecting the other 3 banks in any way.

> The pressure in the operable tanks will not be increased. The operable tanks are credited as maintaining a normal minimum pressure of 3400 psig.
>
> The time limit of 7 days for this Technical Specification is acceptable based on engineering considerations with regard to the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining components and compensatory systems can provide the required capability to maintain the MCRE [ ] habitable. If one bank of tanks is taken out of service, 75 percent of the system will still be available to supply air to the control room in the event of an accident. Dose calculations have been performed to verify that the MCR dose limits will remain within the requirements of GDC 19 if 75 percent (54 hour supply of breathable compressed air) of VES is available and compensatory measures, through the use of the ancillary fans, are taken at 54 hours for the remainder of the event. The MCR ancillary fans are located in the auxiliary building as indicated in Tier 1 Table 2.7.1-5 of the DCD and will be available if required 54 hours after the initiation of VES.
>
> The Tech Spec Bases 3.7.6 as submitted in RAI-SRP6.4-SPCV-06, Revision 0 are marked up to include that GDC 19 requirements are met with 54 hours of VES with compensatory measures taken at 54 hours.

Based on discussions with the staff at a public meeting on December 15, 2009, the applicant revised this response in a letter dated March 25, 2010. RAI-SRP6.4-SPCV-14, Revision 2, included additional required actions in Limiting Condition for Operation (LCO) 3.7.6, Condition D, to provide greater confidence that the MCR envelope can be maintained habitable for 72 hours following a DBA during a condition where one bank of VES emergency air storage tanks is not available. The additional action requires confirmation that the VBS MCR ancillary fans and supporting equipment are available within 24 hours of a bank of VES emergency air tanks going out of service.

Action D.1 has also been clarified. The applicant revised action D.1 to require verification of the pressure in the unaffected banks of air tanks. The pressure in the affected tanks should be verified to be above 23.4 MPa (3,400 psig) within 2 hours and once every 12 hours thereafter. The applicant removed the reference to increasing the pressure in the operable tanks above 23.4 MPa (3,400 psig) to the upper portion of the system operating band from action D.1 and the associated bases. Based on the discussion above and an acceptable finding from the Office of New Reactors Technical Specification Branch, the staff finds the applicant's TS adequate.

6.4.3 Conclusion

The staff has reviewed the applicant's changes to the DCD on the VES system. On the basis of the evaluation described in NUREG-1793 and this report, the staff concludes that the portion of the DCD on the VES system is acceptable and that the application to amend the DC meets the requirements of Subpart B, "Standard Design Certifications," of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," that are applicable and technically relevant to the AP1000 standard plant design. This conclusion is based on acceptable findings by the other responsible NRC branches. Notably, there are related changes in Chapters 3, 7, 9, 15, and 16, which are the responsibility of other branches. Additionally, the DCD revisions reviewed by the staff were submitted as draft.

In a letter dated July 29, 2010, the applicant submitted Change Notice 73. In the letter the applicant identified that the design change noted in this section required a higher pressure at the pressure regulator. As a result, the minimum pressure in the canistered air tanks also needs to be higher. The technical evaluation of this aspect of the design change is included in Chapter 23 of this report.

6.5.2 Containment Spray System

6.5.2.1 Summary of Technical Information

In Revision 17 of the DCD, the applicant changed the description of the nonsafety-related containment spray system in Section 6.5.2.1 and Figure 9.5.1, "Fire Protection System P&ID," in order to correct errors.

6.5.2.2 Evaluation

Revision 15 of DCD Tier 2, Section 6.5.2.1.1, states that the remotely operated valve (FPS-V701) downstream of the manual isolation valve in the spray riser is normally closed, but Figure 9.5.1 shows it as open. In Revision 17 of the DCD, the applicant changed the text to describe this valve as open to match Figure 9.5.1. However, as described in the applicant's January 8, 2009, response to RAI-SRP6.0-SPCV-02, Revision 1, this valve is actually normally closed; therefore, the applicant proposed changes to Section 6.5.2.1.1 and Figure 9.5.1 of the DCD to describe it as such. The staff finds the changes acceptable because the text will revert to the Revision 15 description of the valve and the original error in Figure 9.5.1, which showed the valve as open, will be corrected.

Revision 15 of DCD Tier 2, Section 6.5.2.1.1, incorrectly states that closing the passive containment cooling water system fire header isolation valve (PCS-V005) isolates the primary fire water tank, when it actually isolates the PCCWST. Revision 17 corrects this error, and the text is now consistent with Tier 1, Figure 2.3.4-1, "Fire Protection System," and Figure 6.2.2.1, "PCS P&ID."

6.5.2.3 Conclusion

These changes are acceptable and have no impact on the evaluation of the nonsafety-related containment spray system reported in NUREG-1793, Section 6.5.2.

6.6 Inservice Inspection of Class 2, 3, and MC Components

6.6.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant added metallic containment (Class MC) components to the scope of the preservice inspection (PSI) and inservice inspection (ISI) programs. Furthermore, the changes defined the responsibilities for preparation of the PSI and ISI programs and removed the ISI program discussion from Section 3.8.2, "Steel Containment," which is dedicated to the design, rather than the inspection of the containment.

In Revision 17, the applicant revised Section 6.6.2 to delete the phrases that there are no Quality Group B and C components that require ISI during operation and that relief from Section XI requirements for the baseline DC code will not be required.

6.6.2 Evaluation

In Revision 16 of the AP1000 DCD, the applicant proposed to add Class MC components to the scope of the PSI and ISI programs. 10 CFR 50.55a(b)(2)(vi), “Codes and standards,” provides requirements for licensees to select their effective Edition and Addenda of Subsections IWE and IWL for ASME Code Section XI, as modified and supplemented by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix). ASME Code Section XI, Subsection IWE is dedicated to the PSI and ISI of Class MC components. The AP1000 DCD, Revision 16 changes ensure that the metallic containment integrity is maintained through periodic inspection and testing as defined under the regulations and ASME Code Section XI. In addition, the inclusion of Class MC components as part of the PSI/ISI program provides heightened visibility of the operational program by including Class MC components along with ASME Code Class 2 and 3 components, which receive PSI/ISI. The staff concludes that the change to include Class MC components within the PSI and ISI program is in compliance with the requirements of 10 CFR 50.55a and ASME Code Section XI, and is, therefore, acceptable.

In Revision 17 of the AP1000 DCD, Section 6.6.2, the applicant proposed to remove the phrase that there are no Quality Group B and C components that require ISI during operation.

In the AP1000 DCD, Section 6.6.2, the applicant states that ASME Code Class 2, 3, and MC components are designed so that access is provided in the installed condition for visual, surface and volumetric examinations specified by the ASME Code. Furthermore, it states that considerable experience has been used in designing, locating, and supporting Quality Group B and C (ASME Code Class 2 and 3) pressure-retaining components to permit PSI and ISI. The applicant’s removal of the statement that there are no Quality Group B and C components that require ISI during reactor operation does not eliminate the performance of ISI. Furthermore, neither 10 CFR 50.55a nor ASME Code Section XI prohibits the performance of ISI during plant operation. Since the proposed changes would allow the possibility to expand the performance of ISI during operational conditions rather than eliminate the ISI examinations, the staff concludes that this portion of the proposed change is acceptable.

In Revision 17 of the AP1000 DCD, Section 6.6.2, the applicant proposed to remove the sentence: “Relief from Section XI requirements will not be required for ASME Section III, Class 2, 3, and MC pressure retaining components in the AP1000 plant for the baseline design certification code.”

In the AP1000 DCD, Section 6.6.2, the applicant states that ASME Code Class 2, 3, and MC components are designed so that access is provided in the installed condition for visual, surface and volumetric examinations specified by the ASME Code. Furthermore, it states that considerable experience has been used in designing, locating, and supporting Quality Group B and C (ASME Code Class 2 and 3) pressure-retaining components to permit PSI and ISI. The AP1000 DCD also states that the goal of designing for inspectability is to provide for the inspectability, access and conformance of component design with available inspection equipment and techniques. Factors such as examination requirements, examination techniques, accessibility, component geometry and material selection are used in evaluating component designs, as stated by the applicant.

In addition, the regulations under 10 CFR 50.55a(g)(3)(ii) require that Class 1, 2, and 3 components and supports be designed and provided with access to enable the performance of both PSI and ISI requirements set forth in the Editions and Addenda of Section XI of the ASME

Code incorporated by reference in paragraph (b) of this section applied to the construction of the particular component. However, 10 CFR 50.55a recognizes that inaccessible areas for the containment cannot be eliminated completely. 10 CFR 50.55a(b)(2)(ix)(A) states that for Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. The staff concludes that proposed changes recognize that inaccessible areas may be present in the design of Class MC components and that sufficient effort will be incorporated into the design of Class 1, 2, and 3 components to meet the requirements under 10 CFR 50.55a. The staff concludes that the change is in compliance with the regulations and is, therefore, acceptable.

6.6.3 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 DCD changes as proposed in Revisions 16 and 17 meet the requirements of 10 CFR 50.55a and ASME Code Section XI for Class 2, 3, and MC components and are, therefore, acceptable. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

7. INSTRUMENTATION AND CONTROL

Westinghouse (the applicant) has submitted information in support of its design certification (DC) amendment application that the applicant considers “proprietary” within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5), “Public inspections, exemptions, requests for withholding.” The applicant has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff’s evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears in “square brackets” as follows:

[ ]

The complete text of this chapter, including proprietary information, can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

7.1 Introduction

Chapter 7 of the AP1000 design control document (DCD) includes changes to the descriptions of commitments pertaining to the primary instrumentation and control (I&C) systems of the AP1000 design, as evaluated in NUREG-1793, “Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design,” issued by the NRC in September 2004. Additionally, AP1000 DCD Tier 1, Section 2.5, includes changes to the proposed design description and inspections, tests, analyses, and acceptance criteria (ITAAC) for I&C systems. This report must be used in concert with the original version of NUREG-1793 and Supplement 1 to completely understand the full evaluation of the AP1000 I&C systems standard design. The sections identified and addressed below have had additions, alterations, or deletions incorporated into the technical information presented previously in the certified design of Revision 15 of the AP1000 DCD. The sections not listed below had no appreciable technical changes and, thus, are not included in this report.

Although all open items have been resolved prior to the final issuance of this supplement of the safety evaluation report (SER), this report discusses what open items were generated when it considered an applicant’s original submittal and response to be inadequate.

In the proposed changes to the AP1000 DCD, the applicant provided additional information related to the architecture of its safety-related I&C protection system, referred to as the protection and safety monitoring system (PMS); the diverse back-up system to the PMS, the diverse actuation system (DAS); and data communications protocols and methods utilized to ensure a secure development and operational environment (SDOE). The applicant also proposed minor modifications to some of the interlock and control systems.

7.1.3.1 Compliance with Standard Review Plan (SRP) Criteria

The staff reviewed the additional and amended information provided by the applicant, using the guidance in Chapter 7, “Instrumentation and Control Systems,” of NUREG-0800, “Standard

Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition," Revision 5. The NRC developed NUREG-1793 using the guidance of Revision 4 of NUREG-0800, which did not include Section 7.8, "Diverse Instrumentation and Control Systems," or Section 7.9, "Data Communication Systems." Therefore, although this supplement follows the format of NUREG-1793, the NRC added Section 7.8, "Diverse Instrumentation and Control Systems," and Section 7.9, "Data Communication Systems," to discuss the staff's review of these issues. Where necessary, the staff correlated the information in Sections 7.8 and 7.9 with other pre-existing Chapter 7 sections.

7.1.3.2 Compliance with Industry Standards

The applicant submitted a number of technical reports (TR) associated with I&C systems, which it incorporated by reference into the AP1000 DCD. Based upon the letter, "Secondary References in a Design Certification Rule," dated May 3, 1994, the staff determined with the concurrence of utility representatives and other members of the commercial nuclear industry that documents referenced in Tier 1 and Tier 2 information in the AP1000 DCD should be considered part of the licensing basis. Therefore, the addition of these documents is deemed acceptable to the staff.

In several cases, the applicant referenced revisions of the same industry standards and other regulations in its TRs different from those referenced in DCD Revision 15. The applicant stated that all newly created or revised technical documents that do not refer to the guidance or standards certified in Revision 15 of the AP1000 DCD will reference currently issued guidance or standards provided the newly referenced guidance or standards include all acceptance criteria to those references certified in the Revision 15. The staff found the response acceptable.

7.1.3.3 Compliance with 10 CFR Part 52

Based upon the discussion in Section 7.1.3.3 of NUREG-1793, the standardized power plant designer or combined license (COL) applicant will satisfactorily demonstrate that "the digital I&C system design development process, as documented in the DCD, will ensure that the digital I&C system, as designed, will satisfactorily accomplish its safety functions."

All newly submitted documentation is to support and agree with all previously submitted design documents to enable the NRC to reach the same conclusion as in NUREG-1793.

The regulations in 10 CFR 52.63, "Finality of standard design certifications," are applicable to modifications addressed in this chapter, which supplements NUREG-1793 to address the applicant's amendment request.

7.1.4 Tier 1 Material

Revision 19 of AP1000 DCD Tier 1, Section 2.5, includes information associated with the DAS and the PMS within the first two phases of the software, logic-based, or programmable technology design processes. The proposed alteration to the DAS design allows the use of technology other than microprocessor-based systems (i.e., firmware or analog technology) and the selection of the specific PMS design platform, the Common Q platform. Revision 19 of the DCD also proposes to remove part of the design acceptance criteria (DAC), which is a subset of ITAAC, in Tier 1, Section 2.5. Sections 7.2 and 7.8 of this report discuss these changes.

The applicant classifies the first two phases of the PMS and DAS lifecycles as the design requirements (or conceptual) phase and system definition phase, for both the PMS and the DAS in the certified design. Branch Technical Position (BTP) 7-14, Revision 5 as well as Branch Technical Position HICB-14, to which the AP1000 DCD was certified, in Chapter 7 of NUREG-0800 refers to these two phases of development as the “Planning Activities” phase and the “Requirements Activities” phase of the PMS software lifecycle (SLC) and the DAS programmable technology lifecycle, respectively.

AP1000 DCD Tier 1, Section 2.5.1, identifies the ITAAC for the DAS, and Section 2.5.2 identifies the ITAAC for the PMS. Section 2.5.1, Design Description 4, and Table 2.5.1-4, “Inspections, Tests, Analyses and Acceptance Criteria,” discuss Design Commitment 4, which identifies the phases of the programmable technology development life cycle for the DAS. Likewise, Section 2.5.2, Design Description 11, and Table 2.5.2-8, “Inspections, Tests, Analyses and Acceptance Criteria,” discuss Design Commitment 11, which identifies all phases of the SLC for the PMS. Based on the cover letter received with Revision 17 of the AP1000 DCD, the applicant considers both of these items DAC.

To address the DAC, the applicant provided design information related to the design requirements and system definition phases of the PMS SLC and the DAS programmable technology lifecycle. For both the DAS and PMS sections (Sections 2.5.1 and 2.5.2, respectively) of the Tier 1 document, the applicant proposed the removal of the first two phases of the SLC processes in Revision 17 of the AP1000 DCD. Sections 7.2.5 and 7.2.8 of this report provide further discussion of this topic as it relates to the PMS, and Sections 7.8.2 and 7.8.3 discuss the lifecycle development process for the DAS.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.1.5 Instrumentation and Control System Architecture

In support of the proposed changes to the AP1000 DCD, the staff reviewed the following TRs:

- APP-GW-GLR-071/ Westinghouse Commercial Atomic Power (WCAP)-16675-P, “AP1000 Protection and Safety Monitoring System Architecture Technical Report,” Revision 5 (TR-89). This report describes how the PMS will function. Section 7.2.2, “Protection and Safety Monitoring System Description,” and Section 7.9, “Data Communications Systems,” discuss this report.

- APP-GW-GLR-065/WCAP-16674-P, “AP1000 I&C Data Communication and Manual Control of Safety Systems and Components,” Revision 4 (TR-88). This report provides critical design aspects of the communications methodology and various protocols when dealing with inter- and intra-division communications and safety-related to nonsafety-related communications methods. The report also includes key information related to the manual operation of the AP1000 safety systems. Sections 7.2.2, 7.5.3, 7.9.3, and 7.9.4 of this report discuss WCAP-16674-P in greater detail.

- APP-GW-GLN-022, “AP1000 Standard Combined License Technical Report DAS Platform Technology and Remote Indication Change,” Revision 1 (TR-97), dated May 2007. This report provides information associated with the relocation of DAS equipment, and it also incorporates changes to allow a microprocessor-based or alternative technology to serve as the principal design of the DAS platform. Section 7.1.6, “Diversity and Defense in Depth Assessment of the AP1000 Protection System,” and Section 7.8, “Diverse Instrumentation and Control Systems,” of this report provide additional discussion on APP-GW-GLN-022.

- APP-GW-GLR-017, “AP1000 Standard Combined License Technical Report,” Revision 0 (TR-42). This report summarizes the applicant’s proposed resolution of the 10 generic open items (GOIs) and 14 plant-specific action items (PSAIs) associated with the NRC review of the Westinghouse Common Q platform. Section 7.2.3 includes more information on this document.

- APP-GW-GLR-024/WCAP-16361-P, “AP1000 Setpoint Calculations for Protective Functions,” Revision 1 (TR-28). This report discusses the calculation of setpoints and setpoint methodology for the PMS. Section 7.2.7 addresses additional information on setpoint methodology in AP1000 I&C systems.

- APP-GW-GLR-018, “Failure Modes and Effects Analysis and Software Hazards Analysis for AP1000 Protection System,” Revision 0 (TR-43). This report summarizes the steps taken to perform the failure modes and effects analysis (FMEA) and software hazards analysis (SHA) and serves primarily as a pointer to the AP1000 FMEA and SHA reports.

- APP-GW-JJ-002/WCAP-16438-P, “FMEA of AP1000 Protection and Safety Monitoring System,” Revision 3. This report provides the postulated failure modes and effects the PMS will undergo as a result of the given failures.

- APP-PMS-GER-001/WCAP-16592-P, “Software Hazard Analysis of AP1000 Protection and Safety Monitoring System,” Revision 2. This report discusses the risks associated with the use of software or programmable technology in protection and control systems. The report will be discussed further in Section 7.2.

- APP-GW-GLN-004, “Instrument and Control Design Change,” Revision 0 (TR-39), incorporates signal and other name changes to the post-accident monitoring system (PAMS), which interfaces with the qualified data processing system (QDPS).

- WCAP-17179, “AP1000 Component Interface Module Technical Report,” Revision 2. This report discusses the design of the component interface module (CIM) within the PMS in greater detail than in the WCAP-16675-P report. The report is discussed further in Sections 7.2 and 7.9.

- WCAP-17184-P, “AP1000 Diverse Actuation System Planning and Functional Design Summary Technical Report,” Revision 2. The report defines the planning process and other design attributes of the DAS. The report will be discussed in greater detail in Section 7.8.

- WCAP-17201-P, "AC160 High Speed Link Communication Compliance to DI&C-ISG-04 Staff Positions 9, 12, 13 and 15 Technical Report," Revision 0. This report discusses how the PMS will comply with certain acceptance criteria in the digital instrumentation and controls (DI&C) Interim Staff Guidance (ISG)-04. The TR will be further evaluated in Section 7.9.

- WCAP-17226-P, “Self-Powered Detector Signals in the AP1000 In-Core Instrumentation System", Revision 2. This report discusses the interactions between the nonsafety-related in-core instrumentation system (IIS) and the safety-related core exit thermocouples (CETs). As the report covers the interaction of safety-related CETs and the IIS, this report will be covered in Section 7.5.7.

- APP-GW-J0R-012, “AP1000 Protection and Safety Monitoring System (PMS) Computer Security Plan,” Revision 1. This report discusses how the PMS is constructed in an SDOE and will be discussed further in Section 7.9

- APP-GW-GLR-137, “Bases of Digital Overpower and Overtemperature Delta-T (OP∆T/OT∆T) Reactor Trips,” Revision 1. This report, also known as APP-GW-GLR-005, discusses the changes associated with how the PMS calculates its OP∆T and OT∆T signals. In addition to the content in Section 7.2, Appendix 7A provides additional information regarding this TR.

The staff reviewed WCAP-16592-P, Revision 2, and determined the information in the report adequately addresses the subject matter related to hazards or risks associated with the use of software or programmable technology in the PMS with one notable exception. Revision 1 of the report fails to discuss the potential hazards and/or risks associated with utilizing a firmware-based device, such as the CIM or safety remote node controller (SRNC) within the CIM subsystem of the PMS. While the CIM and SRNC will not use any software during their operation, the use of a programming language and other similar protocols during device development must be addressed and any hazards associated with the CIM and SRNC development process must be adequately mitigated or eliminated prior to their use within the PMS. Based upon the additional information required by the staff to make a determination of acceptability, the applicant submitted a response to request for additional information (RAI)-SRPSHA-01, Revision 1, on June 28, 2010. The staff finds the commitments to update the SHA, including the removal of text stating that since the CIM and SRNC execute no software when operating, there is no need to cover their development or operation in the SHA, are acceptable. Additionally, in the response, the applicant committed to adding several potential hazards and mitigation strategies based upon the use of the CIM system in the PMS in Section 5 of the SHA. The staff previously identified this issue as Open Item OI-SRP7.1-ICE-02. In a subsequent revision to the SHA, the applicant included the potential hazards and mitigation strategies, which resolves this issue.

The applicant replaced the remote shutdown workstation with the remote shutdown room (RSR). The staff finds that the change complies with all applicable acceptance criteria. Specifically, according to design drawings presented to the staff, the RSR transfer switch exists in a hallway located between the main control room (MCR) and the RSR. The staff asked the applicant to explain how it provides positive control of the reactor at all times, in accordance with 10 CFR 50.54(k), “Conditions of licenses.” The applicant responded to RAI-SRP7.1-ICE-13 by stating that control of the RSR switch would be primarily through administrative or procedural control. In a “normal” evacuation of the MCR, licensed operators would pre-staff the RSR before transferring reactor control. In an extreme life-threatening emergency, the licensed operators would be directed to manually scram the reactor before exiting the MCR, and, during the short journey to the RSR, move the transfer switch to RSR control and execute the necessary operator actions that follow a reactor scram. The staff finds this approach acceptable.

In response to RAI-SRP7.1-ICE-14, the applicant provided additional information regarding the overall makeup of its electrical distribution system. The staff reviewed the proposed grounding system for instrumentation and computer systems. Section 8.3.1.1.7 of the AP1000 DCD describes the four primary grounding systems for the AP1000. In particular, the instrument and computer systems use a separate radial grounding system. Based upon the information in the referenced section and the diagrams in Chapter 8 for the Class 1E instrumentation power distribution system, the staff finds the grounded, three-phase wye transformer configuration acceptable for the I&C systems.

In Section 7.1.7, "References," of Revision 17 of the AP1000 DCD, the applicant deleted several references from the certified reference section without sufficient basis. As a result, the staff identified this as Open Item OI-SRP7.1-ICE-03. In its response to Open Item OI-SRP7.1-ICE-03, dated June 22, 2010, the applicant committed to update key reference documents in Chapter 7 of the AP1000 DCD, Tier 2, as well as update Tier 2, Table 1.6-1. Additionally, the applicant committed to update its Tier 2* document list with WCAP-16361-P and WCAP-17179-P, and restore WCAP-15927, Revision 2, to the Tier 2* list as well. In addition, the applicant committed to remove all references to APP-GW-GLR-104, and other associated references to cyber security, as the staff's review under 10 CFR Part 50, "Domestic licensing of production and utilization facilities," does not encompass a cyber security review. Further information on cyber security is provided in Chapter 13 of the SER for the respective COL. Based upon the staff's review of the applicant's commitments in the Open Item OI-SRP7.1-ICE-03 response, the staff finds the changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff could not identify several of the abbreviations the applicant used in some of its originally amended TRs. In RAI-SRP7.1-ICE-25, the staff asked the applicant to define all abbreviations in the text and diagrams in TR-89, and similar documents, as well as terms such as "hardwired," "bypass function," "partial trip mode," and "failsafe trip." The NRC discussed these issues with the applicant's technical staff on January 29-30, 2009, at its Rockville, Maryland office. The staff received adequate responses to RAI-SRP7.1-ICE-24, RAI-SRP7.1-ICE-25, RAI-SRP7.2-ICE-02, RAI-SRP7.2-ICE-03, and RAI-SRP7.2-ICE-06; which the applicant has incorporated into the current revision of WCAP-16675-P and other affected TRs.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.1.6 Defense-in-Depth and Diversity Assessment of the AP1000 Protection System

Section 7.8 of this report discusses the safety evaluation of changes to the approved PRA-based diversity and defense-in-depth (D3) design.

7.2 Reactor Trip System

7.2.2 Protection and Safety Monitoring System Description

As described in Revision 17 of the AP1000 DCD, the applicant identified the Common Q platform as its safety-related protection system platform, negating the reference to WCAP-13383, “AP600 Instrumentation and Control Hardware and Software Design Verification and Validation Process Report,” since that report primarily dealt with the use of the Eagle 21 platform, which will not be used in the AP1000 design. Specifically, in Section 7.1 of the AP1000 DCD Tier 2, the applicant removed references to the use of the Eagle 21 protection system hardware and committed to use the Common Q platform. This change was verified in the applicant’s response to Open Item OI-SRP7.1-ICE-03, dated June 22, 2010, which the staff finds acceptable. As a result, removal of WCAP-13883 from the AP1000 DCD, Tier 2, Chapter 7, is acceptable.

In WCAP-16675-P, Revision 5, the applicant provided design information on the AP1000 PMS. Additionally, the applicant submitted WCAP-16674-P, Revision 4, which addressed data communications and manual component control of safety-related systems, and WCAP-17179-P, Revision 2, which discussed a subsystem within the PMS responsible for allowing nonsafety-related control of safety-related components during normal plant operation known as the CIM system. In the event of a PMS actuation, the PMS overrides nonsafety-related control through the CIM and takes the safety-related components to their safe state. This task is accomplished through priority logic within the CIM. The CIM system resides within the engineering safety features actuation system (ESFAS) portion of the PMS. The CIM system’s primary active components are the CIM and SRNC.

The staff reviewed the design and architecture of the PMS to assess how it meets regulatory requirements and addresses the acceptance criteria associated with an I&C safety system. The PMS encompasses the functions of the reactor trip system (RTS), ESFAS, PAMS, and QDPS. Section 7.5 of this report includes additional information regarding the QDPS.

The discussion of the PMS in the following sections includes information presented in the publicly available, non-proprietary version of the following reports:

- APP-GW-GLR-137, Revision 1 (also known as APP-GW-GLR-005)
- WCAP-16674-NP, “AP1000TM I&C Data Communication and Manual Control of Safety Systems and Components,” Revision 4
- WCAP-16675-NP, “AP IOOOTM Protection and Safety Monitoring System Architecture Technical Report,” Revision 5
- WCAP-17179-NP, Revision 2
- WCAP-16438-NP, APP-GW-JJ-002-NP, “FMEA of AP 1000 Protection and Safety Monitoring System,” Revision 2
- WCAP-16592-NP, Revision 2

7.2.2.1 PMS Functional Requirements

The PMS performs the functions of the RTS, ESFAS, PAMS, and QDPS. During normal operation, administrative procedures and plant control systems maintain the reactor in a safe state, preventing damage to the three barriers (fuel clad, reactor coolant system (RCS), and reactor containment building) that prevent the spread of radioactive material to the environment. Accident conditions causing one or more of the barriers to be threatened could occur. Thus, the PMS monitors key plant parameters and automatically initiates various protective functions to prevent the violation of any of the three barriers. When violation of a barrier cannot be prevented, PMS will attempt to maintain the integrity of the remaining barriers. This ensures that, given a design-basis event, the site boundary radiation releases will be maintained below the limits in 10 CFR Part 100, “Reactor site criteria.” The system functions by actuating a variety of equipment and by monitoring the plant process using a variety of sensors and operations that perform calculations, comparisons, and logic functions, based on those sensor inputs. The PMS functional requirements documents discuss the protective functions of the system and the requirements these functions place on the equipment that performs them.

7.2.2.1.1 Reactor Trip Functions

The PMS generates an automatic reactor trip for the following conditions:

- source range high neutron flux trip
- intermediate range high neutron flux trip
- power range high neutron flux trip (low setpoint)
- power range high neutron flux trip (high setpoint)
- power range high positive flux rate reactor trip
- overtemperature delta-T (OTΔT) reactor trip
- overpower delta-T (OPΔT) reactor trip
- reactor trip on low pressurizer pressure
- reactor trip on low reactor coolant flow
- reactor trip on reactor coolant pump (RCP) underspeed
- RCP bearing water temperature
- pressurizer high-pressure reactor trip
- pressurizer high-water-level reactor trip
- reactor trip on low-water level in any steam generator
- high-2 steam generator water level in any steam generator
- automatic depressurization systems (ADSs) actuation reactor trip
- core makeup tank (CMT) actuation reactor trip
- reactor trip on safeguards actuation
- manual reactor trip

Revision 15 of the AP1000 DCD identified the reactor trip functions listed above. The applicant modified the design description of several reactor trip functions in Revisions 16 and 17 of the AP1000 DCD. Specifically, the applicant proposed the following modifications:

- The source range high neutron flux trip is delayed when the detector’s high-voltage power supply is energized to prevent a spurious trip. The staff finds this acceptable, since the detector would be energized before being declared operable.

- The equations for performing the OTΔT and OPΔT reactor trips were modified. The staff determined, through an examination of TR, APP-GW-GLR-137, Revision 1, and associated RAIs and satisfactory applicant responses, that the new equations were equivalent to the previous equations but in a different format. Therefore, the changes are acceptable. Appendix 7A provides a more detailed evaluation of APP-GW-GLR-137. The revision of the WCAP-16361-P report will be addressed in Section 7.2.7 of this report.

- Reference to Permissive P-8 (power range nuclear power above setpoint) was removed, as Permissive P-10 made it redundant. The staff finds the change acceptable.

- Reactor Trip on High RCP Bearing Water Temperature. The trip has been modified to occur if any RCP experiences a high bearing water temperature without the P-10 interlock being able to block the trip when reactor power is below the P-10 setpoint. As this modification is conservative in nature, in that the number of permissives needing to be satisfied before the trip occurs has been reduced, the change is deemed acceptable.

- The applicant removed the automatic rod withdrawal block, which prevents rod withdrawal if the negative flux rate setpoint is exceeded, in Revision 16 of the AP1000 DCD. As a result, based upon the removal of the associated text and table information in Chapter 7 of Revision 17 of the AP1000 DCD, the staff understands the previously discussed P-17 interlock to have been removed from use in the PMS. Since this control action utilized the nonsafety-related rod control system for its actuation or, in the case of the block signal, the lack of actuation, the removal of the block action is beyond design basis as it pertains to this evaluation and is, therefore, acceptable.

The NRC determined that the other changes to the reactor trip functions, such as editorial changes or changes that added conservatism to the design, were minor; therefore, this chapter does not discuss them further.

7.2.2.1.2 Engineered Safety Features Actuation System Functions

The PMS performs both reactor trip and ESFAS functions. The AP1000 design provides I&C to sense accident situations and initiate engineered safety features (ESFs). The occurrence of a limiting fault, such as a loss-of-coolant accident (LOCA) or a secondary system break, requires a reactor trip plus actuation of one or more of the ESFs. This combination of events prevents or mitigates damage to the core and RCS components and provides containment integrity.

The PMS is actuated when safety system setpoints are reached for selected plant parameters. The selected combination of process parameter setpoint violations is indicative of primary or secondary system boundary challenges. Once the system receives the required logic combination, the PMS equipment sends the signals to actuate appropriate ESF components.

The PMS initiates the following ESF system-level actuations:

- safeguards actuation
- containment isolation
- in-containment refueling water storage tank (IRWST) injection
- CMT injection
- ADS

- RCP trip
- main feedwater isolation
- passive residual heat removal actuation
- turbine trip
- containment recirculation
- steamline isolation
- steam generator blowdown system isolation
- passive containment cooling actuation
- startup feedwater isolation
- boron dilution block
- chemical and volume control system (CVS) isolation
- steam dump control
- MCR isolation
- auxiliary spray and purification line isolation
- containment air filtration isolation
- refueling cavity isolation
- CVS letdown isolation
- pressurizer heater block
- steam generator relief isolation
- normal residual heat removal containment isolation
- demineralized-water transfer and storage system isolation
- reactor vessel head vent valve control

Revision 15 of the AP1000 DCD identified the ESF system-level actuations listed above. The applicant incorporated several minor changes related to the ESF functions into Section 7.3 of the AP1000 DCD, and the staff finds the changes acceptable.

7.2.2.1.3 Component Control Functions

The applicant provides control of individual safety-related components that perform Class 1E functions. Component-level control consists of the following functions:

- resolution of multiple demands (priority logic) for a given component from various systems

- application of manual component demands

- performance of the component protection logic (e.g., torque limit, antipump latch)

- reporting of component status to the plant information system

- local component control

The following inputs are required for control of individual components:

- automatic system-level actuation commands from the reactor trip and ESF actuation logic

- manual system-level actuation commands from the fixed position switches in the MCR and RSR

- individual safety component control commands from the nonsafety-related plant control system (PLS) for component actuations with no onerous consequences (for test, maintenance, restoration, and noncredited actuations)

- individual safety component control commands from the safety and QDPS displays in the MCR for component actuations with onerous consequences

- component feedback signals from the individual safety components to the PMS

The outputs to individual safety-related components consist of hardwired control signals to open or close a solenoid valve, motor-operated valve (MOV), or circuit breaker.

7.2.2.2 AP1000 Protection and Safety Monitoring System Operation

The PMS detects off-nominal conditions and actuates appropriate safety-related functions necessary to achieve and maintain the plant in a safe-shutdown condition. The PMS controls safety-related components in the plant that are operated from the MCR or RSR workstation. In addition, the PMS provides the equipment necessary in its QDPS subsystem to monitor the plant's safety-related functions during and following an accident, as identified in regulatory guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 3.

The AP1000 PMS consists of four redundant divisions, designated A, B, C, and D. The PMS performs the necessary safety-related signal acquisition, calculations, setpoint comparison, coincidence logic, reactor trip and ESF actuation functions, and component control functions to achieve and maintain the plant in a safe shutdown condition. The PMS also contains maintenance and test functions to verify proper operation of the system. The PMS includes four redundant safety displays, one for each division and two QDPS displays, located on the primary dedicated safety panel (PDSP) in the MCR. Only two of the four divisions contain software to drive QDPS displays and to provide PAMS information to the operator, on Divisions B and C. The system's use of four redundant divisions is one of the mechanisms employed to satisfy single-failure criteria and improve system availability.

The I&C equipment performing reactor trip and ESF actuation functions, its related sensors, and the reactor trip switchgear are, for the most part, four-way redundant. This redundancy permits the use of bypass logic, so that a division or an individual channel within a division taken out of service can be accommodated by the operating portions of the protection system that revert to a two-out-of-three (2oo3) logic function from a two-out-of-four (2oo4) logic function. Additional discussion related to bypasses and the indication of inoperable channels within the PMS can be found in Section 7.5.6 of this report.

Four redundant measurements, using four separate sensors, are made for each variable used for reactor trip or ESF actuation within a single division. Each division, which is comprised of two redundant channels, processes one measurement. Analog signals are converted from remote sensor outputs to digital form by analog-to-digital converters within the division's bistable processor logic (BPL) modules - one per channel. Signal conditioning is applied to selected inputs to the BPL, after the digital conversion. Following necessary calculations and processing, the measurements are compared against the applicable setpoint for that variable. A partial trip signal for a parameter is generated in the BPL if the channel's measurement exceeds its predetermined or calculated limit. Processing of variables for a reactor trip is identical in each of the four redundant divisions of the protection system.

Two local coincidence logic (LCL) processors within each division receive the output signal from each division's BPLs. The LCL subsystem acts to initiate a reactor trip or ESF actuation when a predetermined condition in 2oo4 independent safety divisions reaches a partial trip or partial actuation state. The LCL also provides for the bypass of trip or actuation functions to accommodate periodic tests and maintenance.

7.2.2.2.1 Reactor Trip Operation

The reactor trip coincidence logic performs the logic function to combine the partial trip signals from the BPL subsystems and generates a failsafe trip output signal to the reactor trip switchgear. The reactor trip signal from each of the four divisions of the PMS is sent to the division's reactor trip circuit breakers (RTCBs). Each division controls two RTCBs. The design of the RTCB placement ensures that the reactor trip would still occur, even in the presence of a failure of one of the division's two RTCBs. The reactor is tripped when two or more divisions actuate, thereby generating a reactor trip signal opening their breakers. The automatic trip demand signal initiates the following two actions: it deenergizes the undervoltage trip coils on the RTCBs, and it energizes the shunt trip devices on the RTCBs. Either action causes the breakers to trip. Opening the appropriate trip breakers by any two divisions removes power to the rod drive mechanism coils, allowing the rods to fall into the core. This rapid negative reactivity insertion causes the reactor to shut down. Section 7.2.2.3.7, "Capability for Test and Calibration," provides further information on testing RTCBs.

7.2.2.2.2 ESF Actuation

The ESF coincidence logic processors perform the logic function to combine the partial actuation signals from the BPL subsystems, along with automatic and manual permissives, blocks, and resets, to generate a fault tolerant actuation output signal to the integrated logic processor (ILP) subsystems.

The primary functions of the ESF logic processors are to process inputs, calculate system-level actuation, combine the automatic actuation with the manual actuation and manual bypass data, and transmit the data to the ILPs. To perform the ESF coincidence logic calculations, the ESF processors require data from the BPL subsystems and also use manual inputs (such as setpoints and system-level blocks and resets) from the MCR and the RSR workstation.

The ESF logic processors perform the following functions:

- Receive bistable data supplied by the four divisions of BPL subsystems and perform 2oo4 voting on this data.
- Implement system-level logic and transmit the output to the ILP processors for ESF component fan-out and actuation.
- Process manual system-level actuation commands received from the MCR and/or RSR.

The ESF component control function uses redundant ILPs, which serve as LCL output control signal “fan-out” devices, which distribute the activate signal to the various SRNCs that forward their given output to their respective CIMs. The CIMs provide a distributed interface between the safety system and plant equipment for control of non-modulating safety-related plant components. The safety-related input to the CIM comes from both of its respective SRNCs within the PMS, while the nonsafety-related input enters the CIM through the nonsafety-related remote node controller (RNC). Non-modulating control relates to the opening or closing of solenoid valves and solenoid pilot valves and the opening or closing of MOVs and dampers. It also provides the plant operator with information on the equipment status, such as an indication of component position (full closed, full open, valve moving), component control modes (manual, automatic, local, remote), or abnormal operating conditions (power not available, failure detected).

The staff approved the Common Q platform portion of the AP1000 PMS, previously based on information in the Common Q topical report. However, regarding the use of the ESFAS functions, the PMS design uses distribution devices known as ILPs, SRNCs, and CIMs, which had not been previously evaluated in the Common Q platform. The applicant provided design information detailing the functions of each of the aforementioned components. Sections 7.2.2.3.13 through 7.2.2.3.15 include the technical discussion regarding the additional information the staff reviewed for each of these components before determining their acceptability. The ILP is an intra-divisional device that receives its input from the LCL and distributes its outputs to a given SRNCs which forward their outputs to their assigned CIM. Besides distributing the activate signals to non-modulating safety-related devices, the CIM serves as an interface control device, as it receives inputs from both the safety-related PMS and the nonsafety-related PLS to actuate the requested vital components of the safety-related ESFAS.

7.2.2.3 PMS Evaluation

The staff evaluated the technical requirements of the PMS design against the requirements and acceptance criteria in the following documents:

- 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants”

- 10 CFR 50.55a(h), “Codes and standards,” which incorporates by reference Institute of Electrical and Electronic Engineers (IEEE) Standard (Std.) 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations” (NOTE: The clauses of IEEE Std. 603-1991 not referenced in the topical-based discussion below are unaffected by the proposed changes described in Revision 19 of the AP1000 DCD).

- 10 CFR Part 50, Appendix A, “General Design Criteria [GDC] for Nuclear Power Plants”

    – GDC 13, “Instrumentation and Control”
    – GDC 20, “Protection System Functions”
    – GDC 21, “Protection System Reliability and Testability”
    – GDC 22, “Protection System Independence”
    – GDC 23, “Protection System Failure Modes”
    – GDC 24, “Separation of Protection and Control Systems”
    – GDC 29, “Protection Against Anticipated Operational Occurrences”

The industry and staff guidance in Chapter 7 of NUREG-0800 applies to the PMS review.

The staff evaluated the PMS architecture against those requirements affected by the additional or modified design information provided on the PMS. Those requirements include single-failure protection, quality, equipment qualification, system integrity, independence, capability for test and calibration, information displays, control of access, repair, automatic and manual control, and bypasses. Section 7.2.7 discusses setpoints. The staff determined that other requirements for I&C systems were not affected by changes in Revision 19 of the AP1000 DCD, compared to those approved in Revision 15. Specifically, the applicant uses the Common Q platform as a major portion of the PMS. This report evaluated major subcomponents of the PMS, including those that were not previously developed at the time of the staff's original SER for the AP1000 such as the ILP, SRNC, and CIM. Sections 7.2.2.3.13 through 7.2.2.3.15 discuss those evaluations.

7.2.2.3.1 Common Q Evaluation

The staff previously approved the use of the Common Q platform for generic nuclear power plant (NPP) applications. Section 7.2.2.2.2 of this report includes reference material related to the Common Q topical report and the associated safety evaluations. The applicant presented additional planning and design information related to the use of the Common Q platform in the AP1000 design in the form of the newly submitted TRs related to the PMS that are discussed in Section 7.1.5.

7.2.2.3.2 Single-Failure Protection

The staff evaluated the single-failure protection characteristics of the Common Q and PMS against requirements in IEEE Std. 603-1991, Clause 5.1, "Single Failure Criterion," and GDC 21. The staff used the guidance in RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems," which endorses IEEE Std. 379-1988, "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems."

To address the guidance in RG 1.53, the applicant presented the NRC with an FMEA and an SHA to demonstrate the system's capability to withstand a single-failure event. WCAP-16438-P, Revision 2, describes the FMEA, and WCAP-16592-P, Revision 2 describes the SHA for the PMS. The staff's evaluation of the PMS SHA is described in Section 7.1.5, "Instrumentation and Control System Architecture."

The staff examined the AP1000 FMEA against the guidance of IEEE Std. 352-1998, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Safety Systems," and IEEE Std. 379-1988. The staff finds the report provides a satisfactory demonstration of the system's fault tolerance under various scenarios. Several technical questions were identified regarding specific system responses in the presence of certain failure types and required resolution as shown in the following examples:

- Five CIMs possess a mean time between failure (MTBF) that did not meet the design reliability targets for the PMS. However, the text gave no explanation or analysis as to how the system will minimize, mitigate, or eliminate these failure types. To address this issue, the applicant issued a response to RAI-SRP7.1-FMEA-01 in December 2009, stating that an MTBF analysis would be conducted and was expected to be completed by July 2010. The response further discusses how, once the MTBF analysis report is completed, if the failure rates were not in line with the expected failure rate of at least 200,000 hours, the applicant's design change process would be utilized to modify the equipment to reach the MTBF goal or adequately explain why the higher rate of failure is acceptable. The staff finds the use of an MTBF analysis to verify a sufficiently low level of failures of the CIM and SRNC to be acceptable.

- According to the FMEA, when the check sum verification process fails, the setpoints are not updated; however, this condition would be restricted to one division only. The FMEA then discusses how the division may trip erroneously or not trip due to the setpoints not being properly updated. In its response dated December 1, 2009, the applicant stated, in part, that the cyclic redundancy check (CRC) verification mechanism can fail and the correct setpoint value still be monitored by the maintenance and test panel (MTP). Although the MTP may not receive the updates due to a failure of the CRC, there are continuous lower level diagnostics that provide alarm and indication informing the operators that a fault has occurred between the LCL and the MTP. Additionally, even though the diagnostics and self-tests that are executing continuously are not credited, the fault would be limited to a single division's MTP and is, therefore, acceptable.

The FMEA describes how the PMS, through the MTP, periodically adjusts setpoints automatically. This automatic adjustment of setpoints, without operator control, would not meet the requirements in Criterion III, "Design Control," in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." In its response dated November 9, 2009, the applicant related the "periodic refreshing" of data was part of the inherent communications in a deterministic format utilized between the AC160 (Common Q processor) and the MTP. The response stated further that the setpoints within the AC160 can only be altered manually by the operator of the MTP. The applicant submitted Revision 3 of the FMEA to more clearly state that the system is not continually sending "new" updated setpoints to the AC160, but simply performing its communication function. The staff finds this revision acceptable.

When comparing the requirements in Clause 5.1 of IEEE Std. 603-1991 and GDC 21 to the information in WCAP-16675-P and WCAP-16438-P, and after reviewing the responses provided by the applicant related to the FMEA, the staff determined the single failure requirements related to IEEE Std. 603-1991 have been adequately addressed. The NRC staff previously identified these issues as Open Item OI-SRP7.2-ICE-01, and as a result of the staff's review of the responses provided by the applicant and its commitment to update the next revision of the FMEA based upon their responses to RAI-SRP7.1-FMEA-02, RAI-SRP7.1-FMEA-04, RAI-SRP7.1-FMEA-05 and RAI-SRP7.1-FMEA-07 the staff finds the design satisfies the single failure criterion. In Revision 3 to the FMEA, the applicant incorporated the changes described in the RAI responses, which resolves this issue.

7.2.2.3.3 Quality

Clause 5.3 of IEEE Std. 603-1991 and 10 CFR Part 50, Appendix B, require safety-related I&C systems to be designed, manufactured, inspected, installed, and tested under an acceptable quality assurance program. NUREG-0800 Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2"; Section 5.3; and BTP 7-14 specifically address the criteria for a quality software development process. Additionally, the staff evaluated the documentation in the SLC for the Common Q portion of the PMS against the guidance in the following documents:

- RG 1.168, “Verification, Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” which endorses IEEE Std. 1012-1998, “IEEE Standard for Software Verification and Validation,” and IEEE Std. 1028-1997, “IEEE Standard for Software Reviews and Audits”

- RG 1.169 “Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” which endorses IEEE Std. 828-1990, “IEEE Standard for Software Configuration Management Plans”

- RG 1.170, “Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” which endorses IEEE Std. 829-1983, “IEEE Standard For Software Test Documentation”

- RG 1.171, “Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” which endorses IEEE Std. 1008-1987, “IEEE Standard for Software Unit Testing”

- RG 1.172, “Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” which endorses IEEE Std. 830-1993, “IEEE Recommended Practice for Software Requirements Specifications”

- RG 1.173, “Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” which endorses IEEE Std. 1074-1995, “IEEE Standard for Developing Software Lifecycles and Processes”

In Section 7.2.2.2, the staff discusses the approval of the Common Q platform topical report. However, additional requirements were placed on the SLC for the AP1000 safety system, beyond those described in the Software Program Manual (SPM), WCAP-16096-NP-A, “Software Program Manual for Common Q Systems,” Revision 1A, dated January 21, 2005. WCAP-15927, “Design Process for AP1000 Common Q Safety Systems,” Revision 0, placed additional requirements on the design and testing teams developing the PMS. The certified design (Revision 15) of the AP1000 DCD designated the SPM and WCAP-15927 as Tier 2* documents, requiring NRC approval before altering the applicant’s commitments. The staff recently received, reviewed, and approved the use of WCAP-15927, Revision 2. Section 7.2.5 provides further information concerning this matter.

In AP1000 DCD Tier 1, Table 2.5.2-8, Design Description 11, the applicant proposed the removal of design requirements and system definition phases of the PMS SLC ITAAC. The design requirements phase corresponds to the planning activities phase of the SLC in NUREG-0800 BTP 7-14, and the system definition phase corresponds to the requirements activities phase in the same NUREG-0800 BTP. The applicant made available for audit the software planning documents to support removal of the design requirements phase. The staff audited those software planning documents on several occasions, which are discussed below. The applicant describes its quality software development process for the AP1000 project in the Common Q SPM and WCAP-15927, Revision 2, documents. The staff reviewed the proprietary and nonproprietary documentation associated with the first two phases of the SLC as it relates to Common Q portion of PMS system development. Further information regarding lifecycle development and completion as it relates to the PMS is provided in Section 7.2.5.

The applicant originally provided 11 proprietary documents that comprise the design requirements phase of the AP1000 SLC. On April 9-11, 2008; October 9-16, 2008; January 22-30, 2009; and July 30, 2009, the staff conducted site visits at the Westinghouse Twinbrook location in Rockville, Maryland, to review the proprietary documents associated with the design requirements phase. In RAI-SRP7.1-ICE-03, the NRC asked the applicant to explain, in the Tier 2 information of the DCD, how it meets the requirements of the planning ITAAC. This includes a diagram of the planning process and sufficient planning documentation related to the lifecycle development plan for the Common Q and CIM system within the PMS, and the steps taken by the applicant to ensure the PMS is constructed in a SDOE and other project-specific documents. The lack of an adequately detailed CIM system programmable technology lifecycle was captured under Open Item OI-SRP7.2-ICE-05 and the discussion of how the applicant intends to ensure the PMS is constructed in a SDOE will be covered in Section 7.9. The issue of which documents delineate the design requirements or planning activities phases of the PMS developmental lifecycle and their relationship to one another was reconciled in Section 7.2.5 prior to the staff's acceptance of the design requirements phase of the PMS lifecycle development process as complete. As a result, the issue previously identified as Open Item OI-SRP7.2-ICE-02 is considered complete based upon the commitments made in the revised response to Open Item OI-SRP7.2-ICE-02 to include an adequately detailed discussion of the design requirements in Chapter 7 of the DCD. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Several proprietary documents listed as proof of completion of the design requirements phase detail the relationship between two Westinghouse organizations: Repair, Replacement, and Automation Services (RRAS) and Nuclear Power Plants (NPP) (e.g., RRAS AP1000 NuStart I&C Program Project Plan (WNA-PN-00031-GEN[3]) and RRAS AP1000 NuStart I&C Program Project Quality Plan (WNA-PQ-00166-GEN[4])). The documents reveal how the subcontractor (Westinghouse RRAS) interfaces with the parent organization (Westinghouse NPP); however, they originally did not describe how Westinghouse NPP interfaces with, and holds accountable, Westinghouse RRAS, employees, and other subcontractors. The NRC requested this information in RAI-SRP7.1-ICE-04. The applicant provided a written RAI response stating that its RRAS organization will complete all work on safety systems (i.e., all work performed on the I&C safety systems for the AP1000 will be conducted by either RRAS or its subcontractors, who will be required to function under Revision 5 of the Westinghouse Quality Management System (QMS) Manual). The NRC accepted the applicant's use of Revision 5 of the QMS Manual to satisfy the requirements in Appendix B to 10 CFR Part 50. This satisfies RAI-SRP7.1-ICE-04. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In July 2009, the staff conducted an audit of the recently developed proprietary PMS Test Plan, the SRNC and CIM Project Plan, and the CIM System Test Plan documents to ensure compliance with requirements and the applicant's commitments. Overall, the NRC finds the results of the audit to be acceptable. Originally requested under RAI-SRP7.1-ICE-05, the documents listed above satisfy the need for sufficient test plan information related to the planning phase of the Common Q SLC; therefore, the NRC staff considers RAI-SRP7.1-ICE-05 closed.

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[3] Westinghouse altered the numbering format for this document. It is now numbered WNA-PN-00043-WAPP.
[4] Westinghouse altered the numbering format for this document. It is now numbered WNA-PQ-00201-WAPP.

Concerning the proprietary AP1000 NuStart Protection and Safety Monitoring System Software Project Plan (WNA-PJ-00071-GEN), the staff raised issues that included how the applicant could give a value of zero to the overall risk for software development when the risk in some specific areas was rated as high. Furthermore, Appendix A of the same report states that "Test/System Integration Phase Independent Verification and Validation [IV&V] is not within the scope of the AP1000 NuStart Project." The staff requested additional information in this area in RAI-SRP7.1-ICE-08, which discussed the revision to the table. At a meeting in January 2009, the applicant committed to revise notes in the Software Project Plan[5] document, WNA-PJ-00071-GEN, to inform the reader that "zero," in the case of risk, actually means minimal, not "no risk." During the staff visit to the Twinbrook facility on July 30, 2009, the staff confirmed that the applicant had incorporated this item into the proprietary software document. Therefore, this item adequately addresses the RAI, which is now considered closed.

The staff reviewed the proprietary document, WNA-PJ-00071-GEN, "AP1000 NuStart PMS Software Project Plan," and its descendent document, the AP1000/NuStart/DOE Design Finalization and Safety Monitoring System Software Development Plan, WNA-PN-00042-WAPP, as part of the design requirements phase review. Appendix A, Item 4, of WNA-PJ-00071-GEN states that the test/system integration phase IV&V is not within the scope of the AP1000 NuStart project. The software project will be frozen at the processor module software test for Division B software. The staff asked the applicant to clarify these statements; specifically, how the applicant would test Divisions A, C, and D software and the complete software of each division, beyond the processor module software tests. In a letter dated September 3, 2008, "AP1000 Response to Request for Additional Information (SRP7)," the applicant clarified the statements in WNA-PJ-00071-GEN regarding processor module software testing. The letter stated the following:

> The deliverable for the cited plan is a detailed design up to and including the software design. This includes a software demonstration system using a test bed configured for Division B. It was not intended to validate all software for all divisions. The demonstration software for Division B marks the conclusion of the plan's scope.
>
> Once the demonstration software for Division B is complete, a software development plan will be developed to complete the software development life cycle taking credit for work completed in first plan. This would include activities for unit testing, code review, channel integration test for all four divisions, and system integration test, as well as the life cycle V&V activities.
>
> The completed Division B software is a sufficient sample to close the DAC software open issue and the V&V issue, because the completed development of Division B software will demonstrate all of the representative software subroutines included in all of the divisions (i.e., the RPS, ESFAS and QDPS functions).

The staff understands the applicant's explanation that the portion of module testing to be combined and constructed into a test bed mimicking Division B of the PMS in its entirety will

[5] Westinghouse altered the numbering format for this document. It is now numbered WNA-PD-00042-WAPP. Westinghouse also retitled it, "AP1000 NuStart/DOE Design Finalization Protection and Safety Monitoring System Software Development Plan."

serve as a "proof of concept" test for PMS and QDPS software and hardware, rather than an interdivisional integration test of the PMS and QDPS, to which the applicant is committed, later in the SLC process. The staff accepts this approach and closes RAI-SRP7.1-ICE-09.

Section 7.2.5, "PMS Design Process Review," includes a detailed discussion of the level of quality the applicant requires to implement its SLC and programmable technology lifecycle for both the Common Q and CIM subsystem portions of the PMS, and provides the staff's findings as they relate to overall PMS quality.

7.2.2.3.4 Equipment Qualification

Clause 5.4, "Equipment Qualification," of IEEE Std. 603-1991 states, in part, that safety system equipment shall be qualified by type test, previous operating experience, or analyses, or any combination of these three methods, to substantiate that it will be capable of meeting, on a continuing basis, the performance requirements as specified in the design basis. The staff reviewed docketed and Westinghouse proprietary documents relating to the overall methodology and approach when dealing with equipment qualification. The Common Q topical report describes the generic equipment qualification of the Common Q equipment. Additionally, AP1000 DCD Tier 1, Table 2.5.2-8, Items 2, 3, and 4, identify ITAAC to address the seismic, electromagnetic/radio-frequency/electrostatic discharge, and temperature/humidity qualification of PMS equipment. To complete the ITAAC, a licensee incorporating the AP1000 certified design would need to verify that the generic equipment qualification for the Common Q as well as the CIM system portion of the PMS equipment includes the site-specific seismic, electromagnetic/radio-frequency/electrostatic discharge, and temperature/humidity profile. Based on the generic equipment qualification of the Common Q equipment and the ITAAC in Tier 1 of the AP1000 DCD, the staff finds the equipment qualification acceptable for the proposed DC amendment as it relates to the PMS.

7.2.2.3.5 System Integrity

Clause 5.5 of IEEE Std. 603-1991 states, in part, that safety systems shall be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. The staff used the guidance in Appendix 7.1-C and Appendix 7.1-D of NUREG-0800 to review the PMS and its means of meeting the standard in Clause 5.5.

A special concern with any safety-related digital system is confirmation of its ability, in real time, to ensure completion of the protective actions within critical points of time identified in the design basis. Section 7.9 discusses real-time performance with respect to data communications. Items 6a, 6b, and 6c of AP1000 DCD Tier 1, Table 2.5.2-8, would, through their ITAAC, address the overall verification of the system response time. With regard to failure modes, the staff confirmed that the PMS design is such that reactor trip functions fail to the tripped state and ESF functions fail to the as-is state. Additionally, the Common Q equipment contains self-diagnostics to alarm failed conditions and place the system into a fail-safe state. However, the applicant does not propose to use self-diagnostic tests for technical specifications surveillance tests. Based on the ITAAC, FMEA, and self-testing features, the staff finds the PMS design acceptable from a system integrity standpoint.

7.2.2.3.6 Independence

Clause 5.6, "Independence," of IEEE Std. 603-1991 and GDC 22 and GDC 24 require, in part, that safety I&C systems be designed with physical, electrical, functional and communications independence among redundant divisions and between safety and nonsafety systems. The staff compared the independence requirements of the Common Q system to the guidance in NUREG-0800 Appendix 7.1-C and Appendix 7.1-D. Additionally, the staff compared the independence requirements of the PMS as they relate to ensuring the protection system was developed in a SDOE. The evaluation of the system being constructed and operated in a SDOE is discussed in Section 7.9.

Based upon information provided in the Common Q topical report and its appendices, the NRC approved the Common Q platform, which includes the BPLs, the LCLs, the interface and test processor (ITP), and the MTP. While the physical and electrical independence aspects of the Common Q have been demonstrated, several issues remained. One issue concerns the inter-and intra-divisional data communications independence requirements of the Common Q platform. Section 7.9 discusses this issue.

As previously stated, within the layout of the PMS, yet external to the previously approved Common Q platform, the NRC evaluated the RNC, the SRNC, and the CIM against the independence requirements for physical separation and electrical, functional and communications independence (covered in Section 7.9) in Clause 5.6 of IEEE Std. 603-1991. GDC 13, 20, 21, 22, 23, and 24 include other requirements affecting CIM independence.

IEEE Std. 603-1991, which the NRC endorsed in 10 CFR 50.55a(h), defines "associated circuits" as non-Class 1E circuits that are not physically separated or are not electrically isolated by acceptable separation distance, safety class structures, barriers, or isolation devices.

IEEE Std. 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," defines the acceptance criteria for the associated circuits. The RNC, which serves as an interface between the nonsafety-related PLS Ovation® platform and the safety-related CIM, serve as part of the communications isolation function, according to Section 5.2.2 of WCAP-16674-P, Revision 4. This information also appears in Section 3.3.5 of WCAP-16675-P, Revision 5.

According to the description in the TRs, electrical and physical isolation between the RNC and the safety-related CIM is achieved by the use of fiber-optic cabling between the RNC and the remote input/output (I/O) bus, which connects to the PLS side of the RNC. The applicant committed to ensuring that the RNC will undergo qualification testing that meets or exceeds the guidance in IEEE Std. 384-1981. Thus, the device will undergo the qualification requirements placed on Class 1E circuits to ensure that the Class 1E circuits are not degraded below an acceptable level, even in the presence of a failure of the associated circuit. The NRC acknowledges that the RNC will be qualified as an associated circuit, since the fiber-optic cable providing electrical isolation is between the nonsafety-related PLS and the RNC.

The staff understands the subcomponent performing communications conversion is located physically on the CIM and that it creates the functional isolation between the RNC and CIM by performing the communications protocol or language conversion from the Emerson Ovation® programming language to discrete digital signals that will be provided to the priority logic, also located on the CIM. This type of arrangement also creates the functional isolation required to satisfy NRC requirements by ensuring that a failure, degradation, or corruption of the information being received by the CIM will not disable the ability of the PMS to execute or complete its function through the priority module, in accordance with IEEE Std. 603-1991, Clauses 5.2 and 5.6.1. Specifically, the priority logic of the CIM provides the PMS priority over the nonsafety-related control system if a safety-related plant component needs to actuate to its safe state in all cases. This issue was addressed adequately in the applicant's response to RAI-SRP7.0-ICE-03 providing additional information regarding how the PMS has priority over the PLS, even when failure conditions are present. These changes have been incorporated into WCAP-17179-P, Revision 2.

Section 7.9 includes the staff's full evaluation, covering nonsafety-related to safety-related data communications of the PMS, via the RNC and CIM, and other I&C systems.

7.2.2.3.7 Capability for Test and Calibration

The staff evaluated the Common Q system against the requirements of IEEE Std. 603-1991, Clause 5.7, "Capability for Test and Calibration," and the guidance in the following documents:

- NUREG-0800, BTP 7-17, "Guidance on Self-Test and Surveillance Test Provisions"
- NUREG-0800 Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2"
- RG 1.22, "Periodic Testing of Protection System Actuation Functions," issued 1972
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems," issued 1995, which endorses IEEE Std. 338-1987, "IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems"

The staff found that the applicant meets the physical requirements that describe how testing and calibration will be performed administratively, through appropriate procedural guidance for both operational and maintenance personnel, as well as by the design of the Common Q platform and CIM subsystem, which provide various interlocks within the Common Q portion and CIM subsystem within the PMS. Based upon a review of all information in the DCD and referenced TRs, the applicant does not credit any self-testing features with the PMS as replacements for periodic technical specification surveillance tests. The staff finds the applicant's approach acceptable. The staff will continue to verify this finding through the ITAAC process in later development stages (implementation, integration, and factory acceptance testing) of the developmental lifecycles for the Common Q platform and CIM subsystem within the PMS as a whole to ensure the actual system meets the requirements described in the planning activities and requirements activities developmental stages.

The applicant stated that each individual RTCB would be opened during a trip actuation device operational test once per year. In current licensed plants, the maximum length of time between openings of RTCBs (or equivalent) is typically 92 days. This issue is discussed in Chapter 16 of NUREG-1793 and this report. This issue was determined to be adequately addressed in Revision 15 of the DCD and no changes were made to RTCB layout or design, including the periodicity of RTCB testing. Therefore, Open Item OI-SRP7.2-ICE-03 is considered resolved.

7.2.2.3.8 Information Displays

Clause 5.8 of IEEE Std. 603-1991 requires information displays for manually controlled actions, safety-system status, and indication of bypasses. Additionally, the information displays should be accessible to operators. The staff used the guidance in RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems,” issued 1973, during the review.

The information displays used by the PMS remain unchanged since they were found to be acceptable in the “Closeout of Category 1 Open Items for the Common Q Systems,” with the caveat that the flat panel display systems (FPDS) be qualified. Specifically, the NRC’s SER, “Safety Evaluation for the Closeout of Several of the Common Qualified Platform Category 1 Open Items Related to Reports CENPD-396-P, Revision 1, and CE-CES-195, Revision 1” (TAC Number MB0780), dated June 22, 2001, includes the following statement:

> PSAI 6.3 has been resolved generically and may be closed as a requirement in the SE. In connection with its resolution, the staff also found that the FPDS, subject to its successful performance during scheduled qualification testing, to be acceptable for use in Common Q designs as a Class 1E HMI device.

Therefore, a licensee incorporating by reference the AP1000 certified design would need to specifically address the equipment qualification of the FPDS when closing out ITAAC identified in AP1000 DCD Tier 1, Table 2.5.2-8, Design Commitments 2, 3, and 4. The staff finds the AP1000 information display design acceptable as it relates to Clause 5.8 of IEEE Std. 603-1991. Section 7.5 includes further discussion of information displays important to safety. Chapter 18, “Humans Factors Engineering,” evaluates the adequacy of display information presented to operators.

7.2.2.3.9 Control of Access

Clause 5.9, “Control of Access,” of IEEE 603-1991 requires control of access to the PMS. The staff judged the physical capability of the PMS to provide for access control to be acceptable with regard to its system design and administrative controls prior to the removal of the planning activities and requirements activities ITAAC. Section 7.9 includes the evaluation of the ability of the Common Q platform and CIM subsystem within the PMS to meet these requirements to ensure that the PMS is planned and constructed within a SDOE and a protected data communications standpoint. Additionally, Section 7.2.7 discusses the access controls to restrict the alteration of setpoints.

7.2.2.3.10 Repair

The staff compared the information the applicant presented in WCAP-16675-P regarding the capability of the PMS to undergo repair activities both online and offline to meet the requirements of IEEE Std. 603-1991; Clause 5.10, “Repair”; and adequately address the guidance in BTP 7-17. The staff finds that although online self-diagnostics may aid the operator, maintenance, or engineering team to diagnose a fault condition, they do not replace the need for surveillance testing as discussed in Section 7.2.2.3.7.

7.2.2.3.11 Automatic and Manual Control

Clauses 6.1 and 7.1 of IEEE Std. 603-1991 require provisions for automatic control of safety functions. Specifically, Clause 6.1 requires, in part, that means be provided to automatically initiate and control all protective actions except as justified in Clause 4.5. The safety-system design shall be such that the operator is not required to take any action before the time and plant conditions specified in Clause 4.5. Clause 7.1 requires the execute features to receive and act upon automatic controls.

Sections 1.1 and 1.2 of WCAP-16675-P identify the automatic reactor trip and ESF actuations. The applicant did not identify any manual controls for reactor trip or ESF actuations for the PMS in place of automatic controls. Additionally, the reactor trip breakers and ESF equipment are designed to accept and act upon automatic commands from the PMS. Therefore, the staff finds that the PMS design meets Clauses 6.1 and 7.1 of IEEE Std. 603-1991.

Clauses 6.2 and 7.2 of IEEE Std. 603-1991 require provisions for manual actuation of safety functions. Specifically, Clause 6.2 requires, in part, system-level manual actuation of safety functions that minimize the number of discrete operator manipulations and depend on the operation of a minimum amount of equipment, consistent with the constraints of Clause 5.6.1, “Independence Between Redundant Portions of the Safety System.” Position C.4 of RG 1.62, “Manual Initiation of Protective Actions,” issued October 1973, states that equipment common to both the automatic and manual initiation should be kept to a minimum. The action-sequencing logic may be common, as long as component-level control of safety equipment is provided in the control room. Figure 2-2 of WCAP-16675-P identifies the voting logic, as well as the action-sequencing logic, common to the automatic and manual ESF initiation paths. The staff finds that, while this departs from the guidance in RG 1.62, the design is acceptable for the following reason. In the event there is a failure that would disable common portions of the automatic and manual initiation paths in all divisions, a DAS is provided that is not common to the safety-related PMS. The point at which the system-level manual actuation is brought into PMS architecture minimizes the use of equipment and, therefore, reduces the complexity associated with implementing the system-level manual actuation. From a single-failure standpoint, both the automatic and manual initiation paths are designed for single-failure protection. For the RTS-level manual actuation, the switches in the MCR are hardwired to the reactor trip breakers. The staff finds the system-level manual actuation of the reactor trip and ESF to be acceptable.

Section 3.4.2 of WCAP-16675-P describes manual component-level control. The proposed design provides safety-related component-level control of components having onerous consequences, using the soft controls on the safety displays in the MCR and from the CIMs in the equipment rooms. Onerous consequences are defined as those that cause a breach of the RCS pressure boundary or cause a need to shut down the plant to cold conditions to effect repairs. Component-level control of equipment not having onerous consequences is provided through the nonsafety-related workstations in the MCR or the RSR in the event of a condition requiring evacuation of the MCR. The staff finds the proposal to be acceptable, since there is no regulatory requirement for component-level control as it relates to the AP1000 design. Specifically, the plant design does not credit component-level control of equipment to perform safety functions; all safety functions are performed at the system level. The applicant has defined the components having onerous consequences in Table A-1 of the FMEA of AP1000 PMS, WCAP-16438-P, Revision 3, and the staff finds this list acceptable. Although the applicant considers the information proprietary, it will make the component information available to operations, engineering, and maintenance personnel in a licensed power plant using the AP1000 design, thus making the information’s classification acceptable.

7.2.2.3.12 Operational and Maintenance Bypasses

Clauses 6.6, 6.7, 7.4, and 7.5 of IEEE Std. 603-1991 address the requirements as they relate to the operating and maintenance bypass of the sense and command and execute features of the safety I&C system. In part, the clauses require that operating bypasses automatically prevent the activation of an operating bypass or initiate the safety function if the applicable permissive conditions are not met. The capability of a safety system to accomplish its safety function shall be retained while equipment for sense and command and execute features is in maintenance bypass. Additionally, the guidance of RG 1.47 must be addressed to ensure the system responds appropriately in the presence of operational or maintenance bypasses.

The staff evaluated several sections of WCAP-16675-P regarding the PMS bypass capabilities. The four-division system has a high level of redundancy, which allows a division to be bypassed, placing the PMS logic function affecting a reactor trip or ESF actuation into a 2oo3 condition. Additionally, WCAP-16675-P, states that the design does not allow bypassing of two or more redundant channels or divisions. However, placing a given channel into a partial trip condition, if warranted, occurs automatically and does not require the “Function Enable” key switch on the MTP. The AP1000 DCD also commits to the prevention of the operating bypasses when permissive conditions are not met. This aspect of the design would be verified in the ITAAC described in AP1000 DCD Tier 1, Table 2.5.2-8, Design Commitments 9a–d and 11. Thus, based upon the information described above and that found in Section 7.5 of this report, the NRC finds the methodology described in WCAP-16675-P to bypass a given division of the PMS, for maintenance or operational conditions, to be acceptable.

7.2.2.3.13 Integrated Logic Processor Evaluation

The ILPs serve as the action-sequencing logic in the ESFAS portion of the PMS; they distribute the activate signal to the various CIMs, via their respective SRNCs. Based on the design information in WCAP-16675-P, the ILP acts as an intradivisional interface device between the comparative logic device (e.g., LCL) and the SRNC. The SRNC then forwards its output to the priority module (CIM). The ILP uses a design previously approved by the NRC (i.e., Common Q equipment). The staff identified this design information through examination of docketed material in Section 2.2.3.2.2 of WCAP-16775, which revealed the ILP uses an AC160 programmable logic controller to operate. As the AC160 is the processor within the Common Q platform, and the staff has previously approved its use, the staff finds the ILP design acceptable provided the remainder of its components function identically to that of a Common Q based module. The staff previously identified this issue as Open Item OI–SRP–7.2–ICE–04 and, based upon the review of the information in Section 2.2.3.2.2 of WCAP-16775, the staff finds the design of the ILP acceptable.

7.2.2.3.14 Component Interface Module Evaluation

The CIM serves as a transitional device that receives its safety-related input signals from the output of each of its respective SRNCs and delivers its output signal to the respective final actuation device of a given safety-related component. The CIM also serves as an interface device that receives input signals from the nonsafety-related PLS, in addition to the input signal it receives from the SRNC. The nonsafety-related communications (control) signal is applied to the CIM through the nonsafety-related RNC, which serves as a transitional device from the nonsafety-related to the safety-related systems.

The CIM system comprises the following major components:

Component Interface Module	(CIM)
Safety Remote Node Controller	(SRNC)
Double Wide Transition Panel	(DWTP)
Single Wide Transition Panel	(SWTP)
CIM Base Plates	
SRNC Base Plates	

The dual-redundant field programmable gate array (FPGA) based SRNCs receive their [ ] input from their respective ILP within the PMS via a high speed link (HSL). The SRNCs process the PMS signal [ ] prior to the safety-related PMS signal being presented to the CIM. Upon the completion of the signal conditioning being performed by the SRNC, the PMS signal is transmitted to the DWTP, which serves as a connection panel for various signals, and then to the CIM.

The DWTP connects the SRNC base plates and the incoming nonsafety-related Ovation® signal that is processed through the RNC to the input of the respective CIM. The DWTP also serves as the connection point for the 24 volts direct current (Vdc) power feeds that provide power to the SRNC and CIM base plates, the nonsafety-related RNC, and the SWTPs, if any are utilized. The SWTPs serves as extender modules that connect to the DWTP and allow additional CIMs to be connected to the given configuration of CIMs.

The SRNC and CIM base plates serve as physical mounting sockets to which a given SRNC or CIM can be attached.

The output signals from the DWTP or, if utilized, the SWTP, are presented to CIMs, namely the nonsafety-related Ovation® signal from the PLS and the dual redundant, safety-related PMS signal.

The CIM also utilizes FPGA programmable technology, which serves to evaluate and prioritize which of the two signals are present, and position the safety-related component accordingly.

IEEE Std. 603–1991 requires the CIM system satisfy all applicable criteria to ensure the device meets safety system requirements.

To address the single failure criterion in Clause 5.1 of IEEE Std. 603-1991, the applicant submitted WCAP-16438-P, Revision 3, and WCAP-16592-P, Revision 2. The AP1000 FMEA discusses the possibility of individual SRNCs and CIM failure modes and their effect upon the system. Sections 7.1.5, “I&C Architecture,” and 7.2.2.3.2, “Single-Failure Protection,” discuss the findings related to the PMS SHA and FMEA, respectively. The forthcoming FMEA for the CIM system will capture bounding failure types and the anticipated MTBF analysis for the components of the CIM system. Due to the forthcoming CIM system FMEA being bounded by the PMS FMEA (WCAP- 16438-P), the staff finds the approach of addressing the different failure modes of the CIM system adequate. Although both devices do not use software during their operational modes, the use of programming languages along with synthesizers, simulators and other testing tools being used during their lifecycles for both the SRNC and CIM require they be treated as software-based devices. Based upon the updated response to RAI-SRP-SHA-01, Revision 1, dated June 28, 2010, which discusses the CIM system within the SHA, the staff finds the CIM system satisfies the single failure criterion.

IEEE Std. 603–1991, Clause 5.2, requires the CIM system to be designed such that once the system has been actuated the executable features of the protective system continue to completion. Based upon the discussion in the AP1000 PMS FMEA (WCAP-16438-P), upon a failure of the SRNCs, such as through a loss of system power or overt module failure, the device will latch in its last command, such that if a failure occurs, the SRNC will continue to provide the actuate command. In the case of the CIM, should a single communication bus fail on the safety-related bus, the redundant bus will carry the signal through to completion. If a singular CIM fails outright, the system is designed with either a redundant CIM or another system design characteristic, such as a check valve in line with the valve actuator affected by the CIM failure to ensure the system is not adversely impacted by the component failure. These system design characteristics were verified during the April 20-22, 2009 audit conducted at the Westinghouse facility in Cranberry, Pennsylvania.

IEEE Std. 603–1991, Clause 5.3, along with 10 CFR Part 50, Appendix B, requires the CIM system to be designed with a sufficient measure of quality to ensure a high level of component and system reliability consistent with low failure rates. NUREG-0800 Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2"; Section 5.3; and BTP 7-14 specifically address the criteria for a quality software development process. Additional regulatory guidance discussing acceptance criteria related to protective system quality are discussed in Section 7.2.2.3.3, "Quality." The staff found several areas of technical information to be lacking either a sufficient level of detail to satisfy BTP 7-14 or a proposed alternative method of providing adequate CIM system developmental detail in order for the staff to find the CIM development process of sufficient breadth and technical depth. As such, the final disposition of this issue related to CIM development, previously addressed as Open Item OI-SRP7.2-ICE-05, will be covered in Section 7.2.5.

IEEE Std. 603-1991 Clause 5.4, "Equipment Qualification," states, in part, that safety system equipment shall be qualified by type test, previous operating experience, or analysis, or any combination of these three methods, to substantiate that it will be capable of meeting, on a continuing basis, the performance requirements as specified in the design basis. Based upon a review of the ITAAC in Table 2.5.2-8, particularly Design Commitments 2, 3, 4, 5a and 5b, the listed commitments address the protective systems; therefore, the CIM system's ability to withstand the necessary seismic, electromagnetic/radio-frequency, and electrostatic discharge loads or transients. Additionally, the CIMs must be able to withstand additional environmental conditions, such as qualified temperature and humidity profiles. As a result of the requirements placed upon the protective system and, therefore, the CIM system, in the Tier 1 ITAAC of Section 2.5.2 of the AP1000 DCD, the staff finds that the CIM system meets the environmental qualifications as required by IEEE Std. 603-1991.

Safety systems must also be capable of accomplishing their safety functions under the full range of applicable conditions specified in the design basis in accordance with Clause 5.5, "System Integrity," of IEEE Std. 603-1991. Additionally, the staff used the guidance in Appendices 7.1-C and 7.1-D of NUREG-0800 to review the CIM system and the manner by which it meets the requirements of system integrity.

Of particular concern when dealing with computer-based or programmable technology-based systems is the system's ability to complete its function in an actual or real time period that would ensure that any design basis limits are not exceeded. The system must be left in a known state after an event condition has been resolved to satisfy Information Bulletin 80-06; "Engineered Safety Features (ESF) Reset Controls," thus ensuring that it is the operator, not the system that controls the repositioning of equipment during recovery from an event. Additionally, a digital system should be able to detect, alert the operator, and take appropriate action, perhaps by taking a component, channel or division to trip or bypass, once a failure is detected.

Based upon a review of WCAP-16674-P, WCAP-16675-P, and Section 2.3.1.2.8 of WCAP-17179-P, the staff determined the CIM subsystem portion of the PMS system functioned in a deterministically-based manner. This information is also captured in lower level proprietary Westinghouse documents, thus helping to ensure safety-system timing requirements are validated. Based upon Section 6.0, "Component Interface Module," of WCAP-16674-P, the applicant committed that, after the completion of an event scenario, the PMS components will remain in their actuated state until they can be repositioned by an operator. This commitment adequately addresses the guidance of Information Bulletin 80-06. In relation to the guidance of Appendix 7.1-C of NUREG-0800 detailing a protection system's response in the presence of a detected failure of a component, channel or division, the staff finds the information in Section 2.1 of WCAP-17179-P adequately addresses the system's self-identification and reporting of system faults. In addition, the FMEA for the PMS (WCAP-16438-P) discusses the overall system response in the presence of known component, channel or system faults in Section 4.4 ILC, "(Integrated Logic Cabinet) Process Station." Based upon a review of the docketed information presented by the applicant, related to system integrity, the staff concludes the CIM system operates in a manner consistent with Clause 5.5 of IEEE Std. 603-1991.

IEEE Std. 603-1991, Clause 5.6, "Independence," along with 10 CFR Part 50, Appendix A, GDCs 21, 22, and 24, require, in part, that safety-related protection systems be designed with sufficient physical, electrical, functional, and communications independence. The communications independence of the CIM will be discussed in Section 7.9. Electrical isolation of the CIM system is provided through the use of fiber optic connections from the PLS, through the RNC, to the DWTP within the ILC, in which the CIM system resides. Physical independence is provided by housing all CIM system safety-related components separately from the nonsafety-related PLS components, with one notable exception. The RNC is housed in the ILC with the CIM system. Although not classified as a Class 1E component, the applicant committed to qualify the RNC as an associated circuit in accordance with IEEE Std. 384-1981. Although the functionality of the RNC cannot be guaranteed during or after a design basis event, the applicant's commitment to qualify the RNC as an associated circuit prevents the loss of the CIM's safety function in the event of an RNC hardware failure. As such, the interaction of the PLS and PMS within the CIM also addresses functional isolation in that a loss of function within the RNC or another PLS-based component will not inhibit the CIM from carrying out its safety-related function in accordance with 10 CFR Part 50, Appendix A, GDC 24. For further information related to PMS independence, refer to Section 7.2.2.3.6, "Independence."

Based upon a review of the information in WCAP-16674-P, Revision 4; WCAP-16675-P, Revision 5; and WCAP-17179-P, Revision 2, the staff finds that the CIM system meets the criteria related to physical, electrical and functional independence requirements between the safety-related and nonsafety-related systems. A discussion related to the findings on communications independence of the CIM system is provided in Section 7.9.

IEEE Std. 603-1991, Clause 5.7, and 10 CFR Part 50, GDC 21, require the protection system be capable of testing and calibration during all anticipated modes of operation. In addition, the guidance in Appendix 7.1-D and BTP 7-17, of NUREG-0800, as well as RG 1.22, state how a system designed for high reliability will be able to be tested in various configurations while simultaneously ensuring the protection system stays in service. Based upon a review of all applicable information in Chapter 7 of the AP1000 DCD and associated TRs, the applicant credits no self tests that mitigate or replace the need to conduct periodic surveillance testing of the CIM or protection systems. After reviewing the CIM system TR, the system conducts periodic diagnostic self-tests and transmits the information via the CIM and SRNC to the ILP within the PMS, at which point the Common Q portion of the system alerts the operator to any condition outside of normal system operation. This topic is discussed further in Section 2.5 of the CIM TR. The staff finds that based upon a review of the information in WCAP-16674-P, Revision 4; WCAP-16675-P, Revision 5; and WCAP-17179-P, Revision 2; the CIM system satisfies Clause 5.7 of IEEE Std. 603-1991 and 10 CFR Part 50, Appendix A, GDC 21.

Clause 5.9, "Control of Access," in IEEE Std. 603-1991, stipulates appropriate levels of control be required to access the safety system. The PMS is typically accessed from either the PDSP in the MCR or via the division specific MTP. The CIM system components can be accessed when required, such as is the case when safety-related components with non-onerous consequences cannot be controlled via the PLS via the local controls on the CIM itself, but the cabinets in which the CIMs are housed are locked and administrative controls will be in place to prevent unwanted access of the CIM system. A more detailed analysis concerning the control of access to the CIM and PMS is provided in Section 7.9.5, which evaluates the applicant's requirements to ensure that the PMS is constructed in an SDOE.

Based upon a review of the material in WCAP-17179-P, Revision 2, in particular Section 2.12, "Human Factors and Maintenance Considerations," the staff finds that the CIM system adequately addresses Clause 5.10, "Repair," of IEEE Std. 603-1991.

In relation to the use of automatic and manual control, the CIM system is not designed for routine manual control. However, in the event of the failure of a safety-related component with non-onerous consequences that does not have individual safety-related controls on the safety-related PDSP in the MCR (as defined by the PMS FMEA in WCAP-16438-P, Revision 3), the CIM possesses local safety-related controls that can be utilized to position the safety-related component as the situation requires. It should be noted these non-onerous components are typically soft-controlled from the PLS; however, as the PLS is nonsafety-related, its actions are not credited in accordance with IEEE Std. 603-1991. Further discussion regarding how the PMS executes automatic and manual controls may be found in Section 7.2.2.3.11 of this report. Based upon its review of WCAP-17179-P, Revision 2, the staff finds that the CIM system meets the requirements of IEEE Std. 603-1991, Clauses 6.1 and 6.2, "Automatic," and "Manual Control," respectively.

IEEE Std. 603-1991, Clause 5.15, requires the protection system operate in a reliable manner. To facilitate this requirement as it relates to the CIM system, appropriate analyses shall be conducted to ensure that reliability goals have been achieved. The applicant has committed to present to the staff a formal CIM development process, in which it will demonstrate what quality measures, processes, policies, procedures and analyses will be completed in order to satisfy the requirements of IEEE Std. 603-1991, Clause 5.15. Additional discussion of the staff's understanding of the the applicant's commitments made related to CIM system reliability as a byproduct of the CIM development process, may be found in Section 7.2.5 of this report.

The staff conducted an engineering review with the applicant's technical personnel on October 15-16, 2008, and January 29-30, 2009, and conducted two audits, one on April 20-22, 2009, and the other on April 12-16, 2010, to discuss the PMS development process, of which the CIM is a critical part. Previously through the use of Open Item OI-SRP7.2-ICE-05, the staff raised the issue of an inadequate CIM development process and the staff restated this conclusion to the applicant after the April 12-16, 2010 audit and transmitted RAI-SRP7.0-ICE-11 to the applicant on May 17, 2010. After reviewing the updated response to the RAI in WCAP-17179, the staff found that the applicant did not provide sufficient planning or design information related to the CIM priority module. The following provides a topical breakdown of the required information:

- Section 5.1.5 of WCAP-16675-P describes the CIM as a non-software-based Class 1E device that is not considered to be susceptible to a software common-cause failure (CCF). However, the CIM is FPGA-based (programmable technology). The applicant did not provide sufficient information for the staff to determine that the CIM is not susceptible to a software common-cause failure (SWCCF). The applicant committed to updating all TRs by removing all text stating the CIM is not susceptible to a SWCCF.

- The applicant has not adequately described the programmable technology lifecycle development plans and processes for the CIM logic development, design, implementation, testing, and operation.

The applicant has not provided sufficient information on these topics, previously requested under RAI-SRP7.1-ICE-21, RAI-SRP7.1-ICE-22, and RAI-SRP7.1-ICE-23. Additionally, after conducting two audits, one in Scottsdale, Arizona (at the CIM and DAS supplier's facility) and the other in Warrendale, Pennsylvania (Westinghouse's location) and reviewing the information in WCAP-17179, the required information, in both depth and breadth, was determined not to be present for the CIM or its peer components in the AP1000 CIM TR. The staff previously identified this issue as Open Item OI-SRP7.2-ICE-05, and, based upon the applicant's commitment to develop an adequate CIM development process; this open item will be closed out as discussed in Section 7.2.5.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC and, thus, satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.2.2.3.15 Safety-Related Remote Node Controller Evaluation

The staff conducted an audit that dealt with the review of the Phase 1 and Phase 2 proprietary documents for the AP1000 PMS SLC on April 20–22, 2009, in Cranberry, Pennsylvania. During the demonstration of a "test" system, the staff learned of a new device that the applicant would add to the PMS. The applicant demonstrated the use of an SRNC that would serve as the interface device from the ILP to the CIM. Under previous revisions of TRs, no intermediary device existed between the ILP and the CIM. Additionally, the staff reviewed WCAP-17179-P, the AP1000 CIM TR that discussed the use of the SRNC within the CIM system. Based upon the review of the report, questions regarding the use of and quality design process built into the SRNC device remain. Therefore, the issue was captured under the open item related to the CIM. The issue previously identified as Open Item OI-SRP7.2-ICE-06 pertaining to the quality development process of the SRNC will be captured by the response to Open Item OI-SRP7.2-ICE-05, which discusses quality development process of the CIM, of which the SRNC is a part. This open item is discussed in Section 7.2.5.

7.2.3 Common Qualified Platform Design and COL Action Items

In TR-42, the applicant stated its position on what current and future activities it will complete through the ITAAC process. In some cases, the closeout activities either point to more than one ITAAC for closure, or the applicant requested closure of a given PSAI when it had not met all the acceptance criteria for a PSAI. The information in TR-42 is not entirely consistent with the staff's position regarding the disposition of the GOIs and PSAIs.

To identify the applicant's position regarding currently open and previously closed PSAIs and GOIs through the ITAAC process, the NRC asked the applicant, in RAI-SRP7.1-ICE-01, to provide a detailed road map showing which I&C design items remain open and which have already been closed, as well as the method of closure through the ITAAC or DAC process.

On December 12, 2008, the NRC received the applicant's response to RAI-SRP7.1-ICE-01 in DCP/NRC2319. However, several of the applicant's conclusions were not consistent with the staff's findings, which are based on a review of the information in the following documents:

- "Safety Evaluation by the Office of Nuclear Regulation CE Nuclear Power Topical Report CENPD-396-P Common Qualified Platform Project 692," and its supporting future documents regarding the closeout of GOIs and PSAIs, also known as Category 1 and Category 2 Closeout Items

- APP-GW-GLR-017

- Response letter to RAI-SRP7.1-ICE-01, dated December 12, 2008

The staff determined that GOIs 7.1, 7.2, 7.3, 7.5, and 7.6 are closed, based on the findings in "Acceptance of the Changes to Topical Report CENPD-396-P, Revision 1, 'Common Qualified Platform,' and Closeout of Category 2 Open Items (TAC No. MB2553)." The NRC closed out GOIs 7.4 and 7.7 generically, as well as PSAI 6.3. The NRC closed GOI 7.9 and 7.10 in acceptance of "design concept only," based on findings in "Safety Evaluation for the Closeout of Several of the Common Qualified Platform Category 1 Open Items Related to Reports CENPD-396-P, Revision 1, and CE-CES-195, Revision 1 (TAC No. MB0780)."

The following provides the staff's position with regard to the remainder of the open GOIs and PSAIs:

- GOI 7.8

  The staff reviewed WCAP-16674, Revision 4. The staff determined the methodology used by the applicant that the staff understands to be a "safe state, system-based" approach, which typically used the nonsafety-related PLS to control a safety-related component, was acceptable, in that the system first attempts to activate the given safety-related component using the PLS and, failing that, the system then activates using the safety-related PMS. In the event of a SWCCF of the PMS, the AP1000 compensates by activating the DAS to meet the diversity requirements in BTP 7-19. Although other issues related to the development process of the priority device, known as the CIM system in the AP1000 design, still remain, the mechanism by which the safety-related component actuates, either by a nonsafety-related or safety-related system means, is deemed acceptable. Therefore, the staff considers GOI 7.8 closed.

However the issue of overall CIM system quality, planning and development processes continues to be a concern previously captured under Open Item OI-SRP7.2-ICE-05 and, based upon the applicant's commitment to adequately address a high quality CIM development process. This action is discussed in Section 7.2.5.

- GOI 7.9

  The applicant stated in TR-42 that the NRC can close GOI 7.9, regarding the specific use of the ITP and the Advant Fieldbus (AF) 100 buses to provide separation of safety and nonsafety signals, since the AP1000 I&C system differs in some details from the integrated solution described in Appendix 4, "Common Qualified Platform Integrated Solutions." Additionally, this TR provides other plant-specific implementations of safety-to-nonsafety communications.

  Thus, the staff considers GOI 7.9 closed. The staff evaluated the other plant-specific implementations of safety-to-nonsafety communications in Section 7.9.

- GOI 7.10

  The applicant committed to alter the characteristics of the AP1000 Common Q platform so that the AP1000 design no longer uses multichannel operator stations. Therefore, the NRC considers GOI 7.10 closed.

- PSAI 6.1

  The staff agrees with the applicant's conclusion in TR-42 that it uses its quality assurance program to determine the suitability of all I/O devices, by following the requirements of the Westinghouse QMS, which the NRC approved in August 2002. The NRC considers PSAI 6.1 closed.

- PSAI 6.2

  The staff agrees with the applicant's conclusion in TR-42 that, since the AP1000 does not use a hardware user interface, PSAI 6.2 is not valid in the AP1000 specific application. Therefore, the NRC considers PSAI 6.2 closed.

- PSAI 6.3

  The NRC closed out this issue (see above discussion).

- PSAI 6.4

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), as well as the information in "Resolution of Common Q NRC Items for AP 1000," Revision 0, also referred to as TR-42, that this PSAI will not be closed until the completion of the testing phase for hardware and software for the PMS. This commitment to test the system's components to appropriate levels of environmental qualification appears in AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 4.

- PSAI 6.5

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), that this PSAI will be addressed in the testing phase for PMS hardware and software. The commitment to verify the implementation of the SLC appears in AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 11.

- PSAI 6.6

  The staff agrees with the approach the applicant selected regarding the AP1000 setpoint methodology in WCAP-16361-P, Revision 1. AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 10, addresses the actual setpoint accuracy and response time of AP1000 safety systems. Since the DCD provides an acceptable setpoint methodology and ITAAC to verify setpoints and response time, the NRC considers PSAI 6.6 closed. Section 7.2.7 includes further discussion of setpoints.

- PSAI 6.7

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), that this PSAI will be addressed during the human factors engineering (HFE) testing phase for the PMS. AP1000 DCD Tier 1, Chapter 3, ITAAC Table 3.2-1, includes the design commitments for verifying the HFE. Further information is provided in Chapter 18 of NUREG-1793.

- PSAI 6.8

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), that this PSAI is applicable to existing NPPs, not new power plants incorporating new designs. Therefore, the NRC considers this PSAI closed.

- PSAI 6.9

  The staff agrees with the applicant that the plant procedures and/or technical specifications due to installation of the Common Q system will be dealt with at the plant-specific level. Chapter 16 of NUREG-1793 and this supplement address the technical specifications.

- PSAI 6.10

  Previously, the staff reviewed but had not approved the generic FMEA, and submitted several RAIs to the applicant to offer a more detailed technical response or to clarify several statements in the FMEA document that were unclear. Section 7.2.2.3, “Single-Failure Protection,” which discusses the technical information required of the applicant related to its FMEA. As a result, the NRC considered this PSAI open. Based upon the submission of Revision 3 of WCAP-16438-P, the PSAI is considered closed. For further information, regarding this topic, refer to Section 7.2.2.3.2, Single Failure Protection.

- PSAI 6.11

  PSAI 6.11 states that an applicant using the Common Q platform would need to address D3 to prevent an SWCCF. In Sections 7.1 and 7.7 of the AP1000 DCD, the applicant describes the functional requirements of the DAS. The DAC in Revision 19 of AP1000 DCD Tier 1, Table 2.5.1-4, Item 4, includes the design and analysis of the DAS. As described in Section 7.8, the applicant modified Item 4. The staff found that the applicant provided sufficient design information to justify the modifications. Therefore, this PSAI is closed.

- PSAI 6.12

  AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 10, defines the commitment to verify proper response times of circuits. Since the applicant has provided an acceptable ITAAC to address this action item, the NRC considers PSAI 6.12 closed. The time response of the CIM system is addressed in Section 7.2.2.3.14 of this report.

- PSAI 6.13

  The staff agrees that this PSAI will remain open until the completion of integration and preoperational testing. AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 11, includes the commitment to verify load capacity and sharing of communications resources for the PMS.

- PSAI 6.14

  PSAI 6.14 states implementation of the Common Q must not render invalid any previously accomplished Three-Mile Island (TMI) action items. In NUREG-1793, the staff found the AP1000 design addresses the TMI action items. The staff did not find any information in the proposed changes to the AP1000 DCD that would invalidate that conclusion. Since the AP1000 design meets the I&C-related TMI action items described in 10 CFR 50.34(f)(2), “Contents of applications; technical information,” the staff finds this PSAI closed.

Based upon the discussions above, the staff considers all GOIs closed. The open PSAIs listed above will continue to remain open until they are resolved.

7.2.5 Protection and Safety Monitoring System Design Process Review

In AP1000 DCD Tier 1, Section 2.5.2, the applicant describes its entire SLC process related to the development of the Common Q portion of the PMS, which it will implement during the planning, design, construction, testing, and operational phases for the AP1000 I&C safety systems. In AP1000 DCD Tier 1, Section 2.5.2, Design Commitment 11, the applicant deleted the design requirements phase and system definition phase. The applicant based this removal on the cumulative amount of both docketed and audited documentation made available to the staff as of this date.

The staff reviewed the information on the docket and conducted several site visits related to the review of the PMS design process at Westinghouse's Twinbrook facility, and in both Monroeville and Cranberry, Pennsylvania. The primary purpose of the Twinbrook visits (April 8–10, 2008, October 15–16, 2008, January 29–30, 2009, and July 30, 2009) was to conduct an engineering review of the documents for the design requirements and system definitions phases (described as the conceptual phase and system definition phase by the Common Q SPM and WCAP-15927, and the planning activities and requirements activities phases of NUREG-0800 BTP 7-14 SLC Process).

Additionally, the staff conducted three audits, the first was conducted on April 20-22, 2009, examining the Phase 1 and Phase 2 AP1000 PMS SLC proprietary material at the Westinghouse facility in Cranberry, Pennsylvania. The second audit was conducted on March 8–11, 2010, at CS Innovations, in Scottsdale, Arizona. CSI serves as the designer and supplier of the CIM subsystem (as well as the DAS – Section 7.8 provides further discussion of how adequate diversity is maintained between the two systems) within the PMS. The more recent audit was conducted in Warrendale, Pennsylvania at the Westinghouse Automation Services (formerly Repair, Replacement, and Automation Services) facilities on April 12-16, 2010. The more recent of the two audits was in order to determine whether an adequate demonstration of documentation had been presented to conclude that all requirements for the planning activities phase and requirements activities phases for both the PMS and the DAS would be considered complete. The October 3-5, 2006, trip report lists the design requirements phase documents associated with the staff's visit to the Monroeville, Pennsylvania, facility as of October 2006. The applicant based its conclusion that its design requirements and systems definition phases were complete on the proprietary information listed in the April 2009 audit report and the docketed information related to the AP1000 I&C safety systems design process. The applicant desired to close these two phases as part of its DAC closure process.

The staff finds that, once the requirements for each phase of the PMS (SLC) are met, "completion" rather than "elimination" of these and all phases described in the text of AP1000 DCD, Tier 1, Section 2.5.2, Item 11, is appropriate, provided the staff finds the information in those phases to be sufficient. When the staff arrives at that conclusion for each given design process phase, The applicant may remove the given SLC phase(s) in Tier 1, Table 2.5.2-8, Item 11. Those tables describe specific ITAAC activities that will be completed during the given facility's inspection process, rather than the process undertaken to ensure that the applicant has included sufficient quality in the overall design process for AP1000 I&C safety systems. However, the applicant may not remove the design process description from the text-based portion of the design process description in Tier 1, Section 2.5.2, Item 11. The applicant agreed to restore all AP1000 PMS design process phases to the text-based portion of Tier 1, Section 2.5.2, Design Commitment 11. On February 23, 2010, the applicant submitted a response to Open Item OI-SRP7.2-ICE-07 that dealt with the removal of the text based descriptions in Tier 1, Section 2.5.2. The applicant's commitment to reinsert the text-based portions of the PMS design process with the addition of the word "complete" to those given phases, once the staff finds a given developmental phase for the PMS to be adequately addressed, is an approach acceptable to the staff. However, although not specifically addressed by this open item, the staff's expectation would be that this process would be carried out in the Tier 1, Section 2.5.1 for the DAS in a similar fashion. The staff previously identified this as Open Item OI-SRP7.2-ICE-07. In a subsequent revision, the applicant incorporated this Tier 1 information in the AP1000 DCD. The staff concludes that these changes are acceptable.

Based on the review of all audited and docketed documentation provided to the NRC that relates to the design requirements, system definition, and remaining developmental phases of the SLC for the PMS, as well as the series of audits conducted and their conclusions, the staff has not concluded that the applicant adequately completed or addressed the system definition phase, both from a lack of technical adequacy and a lack of completeness of the given subject matter. After reviewing Section 7.1.2.14.1, "Design Process," of the AP1000 DCD, the staff was unable to locate additional information adequately describing all developmental phases of the Common Q SLC and the programmable technology lifecycle for the CIM system in Tier 2 of the AP1000 DCD. Additionally, based upon discussions held during the April 2009, March 2010, and April 2010, audits and conclusions drawn in the respective audit reports, the applicant stated they had "split" the system definition phase of development for the Common Q and CIM system portion of the PMS. However, based upon the documents made available for review, it appears the applicant has "split" the system definition phase of development in a discussion-based format only, as no alteration to the Common Q SPM, the Tier 2* document, WCAP-15927, or Tier 2 information has been made. The applicant did not provide sufficient technical information in the AP1000 DCD, its associated TRs, or its proprietary documentation (to be made available for audit or inspection) to demonstrate satisfactory completion or alteration of the system definition phase of Common Q development.

The applicant must provide adequate information regarding a programmable technology development plan for the CIM and SRNC, previously addressed by Open Items OI-SRP7.2-ICE-05 and OI-SRP7.2-ICE-06, respectively. The applicant must restore the listing of the system definition phase in AP1000 DCD Tier 1, Section 2.5.2, Design Description 11, and in ITAAC Table 2.5.2-8, Design Commitment 11, until the staff finds that the applicant has completed that group of activities. Once those activities had been completed, the staff required the applicant to comply with its commitment as delineated in its response to Open Item OI-SRP7.2-ICE-07, which dealt with the removal and restoration of the text-based descriptions in Tier 1, Section 2.5.2. Additionally, the staff required the applicant to restore information in Tier 1, provide additional, sufficient information in Tier 2, and Tier 2* documentation, especially WCAP-15927, that accurately describes all developmental phases and processes associated with the Common Q portion of the PMS. The staff previously identified this issue as Open Item OI-SRP7.2-ICE-08. Based upon the applicant's commitment to restore the system definition phase of the PMS SLC in a future revision of the AP1000 DCD and based upon the applicant's commitment to add ITAAC Design Description 14, in Tier 1, Section 2.5.2, and Design Commitment 14 in Table 2.5.2-8, in which the applicant describes how it will meet the requirements related to the development of the CIM subsystem within the PMS, the staff determined that Open Items OI-SRP7.2-ICE-05 and OI-SRP7.2-ICE-08 were closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Since the applicant chose to add a more specific ITAAC related to the CIM development process (Tier 1, Section 2.5.2, Design Description 14 and Design Commitment 14 in

Table 2.5.2-8, ITAAC), and based upon the additional language added to the system integration and test phase acceptance criteria in Design Commitment 11, as discussed in Section 7.9.2.3.2 of this report, the staff finds the design requirements, or planning activities phase (in accordance with HICB BTP-14), of development for the PMS to be complete. However, the staff's expectation related to the forthcoming CIM development process is that it will sufficiently address and describe all phases of development for the CIM subsystem of the PMS, including the planning activities phase of development.

Furthermore, since the applicant submitted, and the staff approved, critical licensing documents in the system definition phase related to PMS development (such as the FMEA for the AP1000 PMS (WCAP-16438-P) and the SHA of the AP1000 PMS (WCAP-16592-P)), the staff does not consider the remaining development activities listed as part of the system definition phase of the PMS SLC to be DAC, as future development activities are not anticipated to impact licensing basis information, such as the AP1000 DCD Tier 2 or AP1000 TRs referenced in the DCD. However, if the design detail from future development activities impacts the licensing basis information, the staff expects that information to be incorporated into the licensing basis information.

In Revision 16 of the AP1000 DCD, the applicant asked to remove the reference to WCAP-15927, which the applicant submitted to the NRC in addition to the SPM, to resolve RAI 420.001 and RAI 420.023. The staff issued these RAIs during the certification of the original NUREG-1793 in 2002. The applicant had to demonstrate the measures it would take to ensure critical information in this report is not removed. In July 2009, the applicant decided not to remove the report and submitted Revision 2 of WCAP-15927, which explained which organization, in this case, the separate and independent IV&V organization, has the exclusive responsibility for IV&V activities, including testing activities. The staff considers the issue resolved, since WCAP-15927 elaborates on the organization that performs the software verification and validation (V&V) activities. Specifically, Section 3 of the document states that "…testing activities are defined as part of the V&V process." The statement indicates that the IV&V group is responsible for testing the PMS, as discussed in the Common Q SPM. Therefore, the staff finds that RAI-SRP7.1-ICE-10 is resolved.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC and satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.2.7 Protection Systems Setpoint Methodology

On May 30, 2006, the applicant submitted WCAP-16361-P, Revision 0. The following regulatory requirements and guidance documents apply to the staff's review of WCAP-16361-P:

- GDC 13 and 20

- 10 CFR 50.36(c)(ii)(A), "Technical specifications," requires that the technical specifications include limiting safety-system settings

- RG 1.105, "Setpoint for Safety-Related Instrumentation," describes a method acceptable to the staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and remain within, the safety limit

The Westinghouse setpoint methodology combines the AP1000 uncertainty components to determine the overall channel statistical allowance for the functions of the RTS/ESFAS. All appropriate and applicable uncertainties, as defined by a review of the AP1000 baseline design input documentation, have been considered for each RTS/ESFAS function. The methodology used to combine the uncertainty components for a channel is an appropriate combination of those groups that are statistically and functionally independent. Those uncertainties that are not independent are conservatively treated by arithmetic summation and then systematically combined with the independent terms. It includes instrument (sensor and process rack) uncertainties and non-instrument-related effects (process measurement accuracy). The methodology used the square-root-of-the-sum-of-the-squares technique, which the NRC has approved. Also, the American National Standards Institute (ANSI), the American Nuclear Society (ANS), and the International Society of Automation (ISA) approve the use of the same probabilistic and statistical techniques for the various standards that determine safety-related setpoints.

The staff reviewed WCAP-16361-P and found that the allowable values (AVs) are equal to the rack calibration accuracy, which is the acceptable "as-left" value. This methodology ensures that the purpose of the AV is satisfied by providing a large enough allowance to account for those uncertainties not measured during the surveillance tests to protect the safety limit. Also, the difference between the AV and the nominal trip setpoint is as large as the calibration tolerance, and the AVs, along with the nominal trip setpoint, are included in the plant technical specifications as the associated criteria, in accordance with 10 CFR 50.36. Therefore, the staff concludes that the proposed WCAP-16361-P, Revision 0, as it realtes to an overall setpoint methodology, is acceptable. However, due to proposed changes in the inputs to the OPΔT and OTΔT as discussed in Section 7.2.2.1.1 of this report, the applicant committed to revise the WCAP-16361-P report to address these changes. These changes were incorporated into Revision 1 to WCAP-16361-P, which the staff reviewed and found acceptable.

Section 7.1.6.1 of the AP1000 DCD states that all requested information on the subject of setpoint methodology and final setpoint calculations has been completely addressed and requires no further action by the COL applicant. This statement is not in agreement with WCAP-16361-P, in which the applicant concludes that it cannot determine the final setpoint calculations until it completes the final design of the power plant. The staff issued RAI-SRP7.2-ICE-08, requesting that the applicant demonstrate how it intends to meet the final calculation requirements, given that it has not completed the protection system design. The applicant submitted DCR/NRC2315 to the NRC on December 9, 2008, declaring that the COL applicant will determine the setpoint adequacy, in accordance with AP1000 DCD Tier 1, Table 2.5.2-8, Item #10. In its response to Open Item OI-SRP7.2-ICE-09, dated March 8, 2010, the applicant committed to restore the language in Section 7.1.6.1, stating that the COL applicant or licensee will be responsible for the final determination of setpoints, in accordance with AP1000 DCD Tier 1, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 10. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.2.6 of WCAP-16675-P states that the applicant uses the MTP to alter setpoints and addressable constants. The MTP provides a dedicated display interface for each division and is used to bypass the channel before changes are made. The staff evaluated the access controls on the MTP to control the alteration of addressable constants, setpoints, parameters, and other settings to meet the requirements of IEEE Std. 603-1991, Clause 5.9. This clause required the design to provide administrative control of access to safety-system equipment. Section 1, Point 10, of the digital I&C ISG Document #4, "Highly-Integrated Control Rooms—

Communications Issues (ISG #4-HICRc)," Revision 1, clarifies this requirement by making the following statement:

> A workstation (e.g., engineer or programmer station) may alter addressable constants, setpoints, parameters, and other settings associated with a safety function only by way of the dual-processor/shared-memory scheme described in this guidance, or when the associated channel is inoperable. Such a workstation should be physically restricted from making changes in more than one division at a time. The restriction should be by means of physical cable disconnect, or by means of keylock switch that either physically opens the data transmission circuit or interrupts the connection by means of hardwired logic.

The staff finds the use of the MTP acceptable to bypass the channel in addressing the guidance in ISG No. 4-HICRc. Specifically, since the MTP serves as a dedicated display interface system for each division, the NRC does not require physical means to prevent the MTP from making changes to more than one division at a time.

7.2.8 Protection and Safety Monitoring System Evaluation Findings and Conclusions

The staff evaluated the revisions made to Chapter 7 of the AP1000 DCD, against the regulatory requirements stipulated in NUREG-0800 Table 7-1. Below is a summary of the staff's findings as they relate to the AP1000 PMS.

The staff finds that the proposed changes to the AP1000 DCD meets the requirements of 10 CFR 50.55a(a)(1). Specifically, the staff found that the applicant incorporated quality standards into the design of the AP1000 I&C systems.

Regulations in 10 CFR 50.55a(h) require compliance with IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. The staff compared the PMS in the amendments to the AP1000 I&C systems design with the applicable clauses of IEEE Std. 603-1991 and has the following findings:

- The applicant satisfied Clause 5.1 of IEEE Std. 603-1991, as documented in Section 7.2.2.3.1.1. Specifically, the applicant has demonstrated how the PMS met the criteria presented in Clause 5.1 and GDC 21. The FMEA provided to the staff adequately demonstrates how the PMS will operate with a single failure under all postulated operating conditions, as discussed in Section 7.2.2.3.2 of this report.

- IEEE Std. 603-1991, Clause 5.3: The applicant satisfies the requirement of quality for the PMS, as documented in Section 7.2.2.3.3. Specifically, the applicant has not demonstrated how it meets the criteria in Clause 5.3 with regard to the design of the CIM and the SRNC, which would then provide reasonable assurance that it had developed a high-quality product for all components within the PMS. Sections 7.2.2.3.14 and 7.2.2.3.15 of this report discuss these issues. The staff previously identified these issues as Open Items OI-SRP7.2-ICE-05 and OI-SRP7.2-ICE-06, respectively. Based upon the applicant's commitment to add an ITAAC relating to the development of a programmable technology lifecycle for the CIM system, as demonstrated by the addition of Design Description 14 and Design Commitment 14 to Table 2.5.2-8 of the DCD, that is consistent with the requirements of IEEE Std. 603-1991 and the guidance of BTP 7-14 of NUREG-0800, the staff considers the open items closed.

10 CFR Part 52, “Licenses, certifications, and approvals for nuclear power plants,” Appendix D, “Design Certification Rule for the AP1000 Design,” Section II.D, defines Tier 1 information as that information explaining design descriptions, along with ITAAC information. Based upon the applicant’s commitment to restore all phases of the PMS SLC design process in the text-based portion of Item 11 in NUREG-0800 Section 2.5.2, “Protection and Safety Monitoring System,” the issue the staff previously identified as Open Item OI-SRP7.2-ICE-07 is closed.

- 10 CFR 52.47(b)(1) describes the ITAAC. The applicant provided sufficient information to satisfy the completion of the design requirements or conceptual phase of the PMS SLC, with the commitment to add the CIM Development Process ITAAC discussed above. Based upon its review of the documentation presented and based upon conclusions drawn in several audits, the staff will not approve the removal of the system definition phase of the PMS SLC until such time as the applicant provides satisfactory information to the staff for review and approval.

- IEEE Std. 603-1991, Clause 6.8: The staff found the proposed setpoint methodology acceptable. Additionally, the applicant commitment to restore the text in Section 7.1.6.1 of the AP1000 DCD stating that the COL applicant or licensee will be responsible for the final determination of setpoints was found acceptable to the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff finds that due to the addition of the ITACC related to the information that will be provided for the CIM development process, (design description and Design Commitment 14 in Tier 1, Chapter 2.5.2 and Table 2.5.2-8, respectively) the design requirements or planning activities phase of development for the PMS is considered complete.

The staff finds that the proposed changes to the AP1000 DCD did not affect the remaining requirements of IEEE Std. 603-1991. Therefore, the staff’s original conclusions in NUREG-1793 for these requirements are still valid.

The staff finds that proposed changes to the AP1000 DCD did not present changes that would invalidate the staff’s conclusions in NUREG-1793 regarding the requirements in 10 CFR 50.34(f)(2).

Appendix A to 10 CFR Part 50 provides specific criteria for the I&C systems. The staff found that the PMS design continues to comply with the specific GDC for the I&C systems in NUREG-0800 Table 7-1, as described in NUREG-1793.

Regulations in 10 CFR 52.47(a)(9), “Contents of applications; technical information,” require that applications for light-water-cooled NPPs evaluate the standard plant design against NUREG-0800 revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in NUREG-0800 acceptance criteria. NUREG-0800 Chapter 7 provides the design considerations for safety, including criteria for performance and reliability considerations. The staff evaluated the information presented for AP1000 safety systems in the AP1000 DCD against the guidance provided in NUREG-0800 Chapter 7. With the exception of the items listed above, the staff finds the PMS design descriptions to be acceptable.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.3 Engineered Safety Features Actuation Systems

7.3.1.4.1 Automatic Depressurization System Valve Block

The AP1000 design uses ADS valves in certain accident conditions to depressurize the RCS and allow passive safety systems to inject coolant to the reactor core. However, a potential CCF of the PMS could initiate the ADS and simultaneously cause the PMS to not appropriately respond to the condition. As a result, the plant would need to respond to the spurious actuation of the ADS valves using back-up systems, such as the DAS, nonsafety systems, and operator action. While the scenario is beyond design basis, the consequences of such a scenario would be similar to that of up to a large-break LOCA. In addition, the staff did not identify analyses demonstrating the capability of the plant to mitigate the scenario of a spurious actuation of the ADS valves.

To address the potential spurious actuation of ADS valves due to a CCF of the PMS, the applicant added an ADS valve block feature to prevent the spurious actuation of any of the ADS valve paths in each of the four stages of valve sets. The blocking feature prevents one valve in each of the two-valve valve sets from opening without deactivation of the block signal. For ADS Stages 1-3, the blocking feature is applied to the depressurization valve, and regarding ADS Stage 4, the blocking feature is applied to the squib valve. For ADS Stage 4, the applicant will determine whether the “arm” or “fire” signal within the squib valve will have the blocking feature applied. However, based upon the information provided, the staff determineds the use of either signal is acceptable.

Section 7.3.1.2.4.1 of the AP1000 DCD describes the need for a confirmatory process condition that is separate from the PMS actuation logic to serve as the input to the blocking signal circuit. The DCD describes the use of redundant CMT level switches, one in each CMT, to serve as the permissive that removes the block signal. Each of the respective level switches act to clear the block signal upon a lowering level in a given CMT once the tank level has reached or surpassed its setpoint. The DCD further states the blocking device will be a Class 1E module that will satisfy the requirements of safety-related I&C equipment. On a divisional basis, the interface between PMS and the blocking device will be the CIM for the each affected ADS valve in the division. The CIM resides in the PMS circuitry after the logic function has taken place. Additionally, the AP1000 DCD states that there are no interdivisional connections between the blocking devices nor will there be any coincidence voting between the blocking devices thus satisfying the independence requirements of IEEE Std. 603-1991. Section 7.3.1.2.4.1 of the AP1000 DCD also discusses the use of manual switches to enable the operator to manually clear the block signal as required. Since this action affects a component with onerous consequences the staff expects, that beyond the commitment in the AP1000 DCD for this switch to reside in the MCR, the given switches will be separate, hardwired switches in the primary dedicated safety panel or another safety-related panel in the MCR. In addition, the applicant updated Figure 7.2-1 noting that an ADS valve block signal is utilized via the CMT level switches’ signal.

Beyond the redundancy created through the use of two CMT level switches to prevent a single level switch from causing a blocking circuit failure as described in the AP1000 DCD, in accordance with the description in the FMEA for the AP1000 PMS, WCAP-16438-P, the text explains that should an ADS block signal within a division fail to remove the block signal, only a single division of PMS is affected. For Stages 1-3 ADS valves the other division's ADS valves will actuate and as Stage 4 ADS valves utilize signals from two divisions, the "other" PMS division's signal will actuate the given Stage 4 ADS valve.

Based upon a review of the information related to the ADS valve blocking circuit provided in the AP1000 DCD and the AP1000 FMEA for the PMS, the staff finds the addition of the ADS valve block circuit to be acceptable. In a subsequent revision to the AP1000 DCD and the FMEA, the applicant incorporated these changes, which resolves this issue.

7.3.4 ESFAS Evaluation Findings and Conclusions

The ESFAS is a portion of the PMS. AP1000 DCD, Section 7.2 discusses the additional design information associated with the PMS and the staff's evaluation. The staff identified no changes of substance in Section 7.3 of the AP1000 DCD other than those described below.

To prevent spurious actuation in of any of the valve paths for any and all of the stages of the ADS valves, the applicant chose to implement an ADS valve blocking circuit that prevents the given depressurization valve (for Stages 1-3 ADS valves) or the squib valve (for Stage 4 ADS valves) from opening unless system conditions warrant. Based upon a review of the information in AP1000 DCD, Section 7.3.1.2.4.1 and the FMEA for the AP1000 PMS, WCAP-16438-P, the staff finds the addition of the ADS valve block circuit to be acceptable These changes were incorporated in Revision 19 to the AP1000 DCD and WCAP-16438-P, Revision 3, which resolves this issue.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.4 Systems Required for Safe Shutdown

7.4.3 Evaluation Findings and Conclusions

The applicant changed the fifth bullet in Section 7.4.3.1.3 of the AP1000 DCD to clarify that the remote shutdown workstation is designed, not for a single failure, but with redundancy. However, when a random event other than fire causes an MCR evacuation, a coincident single failure in the safety systems controlled from the remote shutdown workstation is considered.

The staff finds the change acceptable, since RG 1.189, "Fire Protection for Operating Nuclear Power Plants," establishes the bases for safe shutdown with respect to fire protection. These limits do not require consideration of an additional, random, single failure in the evaluation of the capability to safely shut down because of fire. NUREG-0800 Section 9.5.1 addresses conformance to RG 1.189. Therefore, the application of the single-failure criterion to remote shutdown is applicable only to other events that could cause the control room to become uninhabitable. These events would not result in consequential damage or unavailability of systems required for safe shutdown. The AP1000 design does consider events other than fire, coincident with a single failure in the safety system, for the remote shutdown workstation.

7.5 Safety-Related Display Information

7.5.3 Network Gateway (Real Time to Protection and Safety Monitoring System)

The applicant removed the communications description from Revision 17 of AP1000 DCD Tier 2, Section 7.1.2.8, and added a reference to Section 3 of WCAP-16675-P, Revision 3. WCAP-16675-P, as supplemented by WCAP-16674-P, Revision 2 provides a comprehensive description of the communications within the safety system, between safety and nonsafety systems, and within the nonsafety systems. Section 7.9 documents the review of the modifications made to the AP1000 safety and nonsafety communications system, and provides additional information on the communications system.

7.5.5 Qualified Data Processing System

Revision 17 of the AP1000 DCD upgraded several variables in Table 7.5-1, "Post-Accident Monitoring System," to add seismic qualification to some instruments and to add a QDPS indication. The staff accepts this change, which increases safety with more highly qualified instruments and controls, as well as improved information to support MCR operations.

Revision 17 of the AP1000 DCD changed the information given for several variables in Table 7.5-1 and added Note 7 indicating, "This instrument is not required after 24 hours."

The staff finds the addition of Note 7 acceptable for these variables because AP1000 DCD, Section 7.5.4, includes the statement below:

> Class 1E position indication signals for valves and electrical breakers may be powered by an electrical division with 24-hour battery capacity. This is necessary to make full use of all four Class 1E electrical divisions to enhance fire separation criteria. The power associated with the actuation signal for each of these valves or electrical breakers is provided by an electrical division with 24-hour battery capacity, so there is no need to provide position indication beyond this period. The operator will verify that the valves or electrical breakers have achieved the proper position for long-term stable plant operation before position indication is lost. Once the position indication is lost, there is no need for further monitoring since the operator does not have any remote capability for changing the position of these components.

7.5.5.1 Combined License Information and Tables 7.5-1 and 7.5-8

Section 7.5.5 of the AP1000 DCD states: "This section has no requirement for information to be provided in support of the Combined License (COL) application." Section 7.5, Tables 7.5-1 and 7.5-8, indicate that the meteorological parameters and environs radiation and radioactivity variables are "site specific." The staff requested clarification of this inconsistency in RAI-SRP7.5-ICE-02. The applicant's response states: "The words 'site specific' for environs radiation and radioactivity parameters indicate that these variables are site-related and are to be addressed by the COL applicant in the site emergency response plan identified in DCD Tier 2, Chapter 13, Section 13.3.1. Therefore, each COL applicant is to provide information for monitoring the meteorological parameters and environs radiation and radioactivity as appropriate." In a letter dated May 26, 2010, the applicant provided a revised response to Open Item OI-SRP7.5-ICE-01 stating that the applicant commits to update Tier 2 Tables 1.8-1 and 1.8-2 in Revision 19 to the DCD. The applicant plans to modify Section 7.5.5 to indicate a

COL action item regarding meteorological parameters and environs radiation and radioactivity instrumentation is required by the COL applicant. The staff previously identified this as Open Item OI-SRP7.5-ICE-01 and, based upon the commitment in the applicant's response to the open item, the open item is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

7.5.6 Bypass and Inoperable Status Information

The applicant removed the description of the bypass and partial trip in AP1000 DCD Tier 2, Section 7.1.2.9, and provided a reference to Section 6 of WCAP-16675-P, Revision 5. Section 6.4 of WCAP-16675-P describes the design requirements for the bypass and partial trip conditions.

This section establishes the use of the Common Q network to provide bypass status indication in the MCR, in accordance with RG 1.47.

<u>Evaluation</u>

Regulations in 10 CFR 50.34(f)(2)(v) require the safety-system design to provide an automatic indication of the bypassed and inoperable status of safety systems. As applied to data communications systems, NUREG-0800 Section 7.9 states that the bypassed and inoperable status indications for data communications systems should be consistent with those of the systems of which they are part. The bypassed and inoperable status indication should conform to the guidelines of RG 1.47.

The staff finds the AP1000 DCD, as supplemented by WCAP-16675, sufficiently demonstrated how the PMS design provides bypass indications of protection channels used in the reactor trip and/or ESF actuation path to meet the requirements of 10 CFR 50.34(f)(2)(v). Specifically, the staff concludes the applicant provided adequate information that describes how the indications of bypassed channels or components on the MTP and MCR conform to RG 1.47, as specified in the guidance provided in NUREG-0800 Section 7.9. The staff previously identified this issue as Open Item OI-SRP-7.5-ICE-02, and based upon the information in Section 6.4, "Bypass and Partial Trip Condition," in WCAP-16675, the staff finds the design of the system's ability to inform the operators of bypassed and inoperable safety channels within the PMS to be acceptable.

7.5.7 In-Core Instrumentation System

7.5.7.1 In-Core Instrumentation Interaction with Core Exit Thermocouples

WCAP-17226, "Assessment of Potential Interaction between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000™ In-core Instrumentation System," Revision 2, describes how the AP1000 IIS design satisfies the requirements of IEEE Std. 384-1981 such that any credible single fault in the non-Class 1E self-powered detector (SPD) signals will not reduce the number of valid Class 1E CET inputs to the PAMS below the required minimum number. The CET signal wires used by the Class 1E PAMS and the non-Class 1E SPD signal wires used by the on-line power distribution monitoring system (OPDMS) are in very close proximity, and do not satisfy the separation distances identified in IEEE Std. 384-1981. In addition, the required separation distance identified in IEEE Std. 384-1981 between safety and nonsafety signals is not met in the in-core instrument thimble assemblies (IITAs) and in the mineral insulated (MI) cable assembly hardware that route the

CET and SPD signals from the reactor vessel head (RVH) penetrations to the refueling disconnect panel (RDP). Furthermore, four of the AP1000 IITA contain non-Class 1E CET sensors that provide input to the non-Class 1E DAS.

Within the IITA, the active portions of the Class 1E CET elements and the non-Class 1E SPD elements are placed inside individual steel outer sheaths that share a common ground to provide electrical isolation between the CET and SPD elements. The presence of two commonly grounded metallic barriers within the IITA probe assembly and in the MI cables makes it incredible for an SPD emitter signal to short directly to the CET element signal leads.

From the Quickloc flanges on the RVH to the RDP, the CET signals and SPD signals share a common Class 1E design and post-accident environmentally qualified MI cable assemblies. The MI cable assemblies consist of individual flexible steel-sheathed cable sub-assemblies, which contain the separate CET and SPD signal lead MI cables, routed together in a flexible steel outer sheath that serve as conduits for individual CET and SPD cables in the cable assembly.

From the Class 1E connector panels on the RDP, the SPD signals are split and sent to two signal processing system (SPS) cabinets that are powered by non-Class 1E power supplies. The CET signals are also split at the RDP Class 1E connector panels, where most of the CET signals are sent to the PMS; the remaining 4 CET signals are sent to the DAS. The CET signals sent to the PMS are split into two corresponding trains and sent to the corresponding PMS divisions via separate qualified Class 1E MI cables. The CET signals sent to the DAS are routed through post-accident environmentally qualified MI cables.

An analysis was performed to determine whether the non-Class 1E power supplies that power the SPS cabinets can have an over-voltage or surge voltage that propagate backwards to the SPD inputs signals through the SPS circuitry without attenuation or shorting to ground. The analysis showed that it is credible that a sufficiently large over-voltage or a voltage surge at the SPS cabinet power supply inputs could cause at least a momentary loss of all Class 1E CET signals associated with the affected SPS cabinets via shorting between the SPD and CET wires in the backshell of the IITA or MI cable electrical connectors. If the over-voltage or transient surge condition were to occur on both SPS cabinets, then the result could be that all of the CET signals needed by the PAMS become inoperable.

To mitigate this concern, the applicant performed an analysis to determine the maximum sustained over-voltage value for low voltage circuits in Westinghouse NPP designs that run cable in accordance with nuclear industry standards. The staff requested the applicant describe how the maximum credible over-voltage generated by the SPS cabinets is established. In its response, the applicant proposed to modify WCAP-17226 with the following explanation:

> AP1000 requires that low voltage systems be installed in a separate raceway system from medium voltage systems. As such, the maximum credible sustained overvoltage condition which can occur in a low voltage power or control circuit routed in this (these) low voltage raceway system(s) can be determined conservatively by considering nominal system operating voltages and maximum preferred system voltage range as defined in ANSI C84.1-2006. The system voltage at the low voltage system will remain balanced when the medium voltage system is supplied from normal or reserved source of power during the normal plant operation. During the abnormal plant operation when the normal and reserve sources of power are not available, the low voltage system will

continue to function by receiving power from the standby diesel generators. The system voltage will also remain balanced even when the medium voltage continues to operate in the presence of a single line to ground fault indefinitely.

As the neutral of the load center transformers secondary windings are solidly (or effectively) grounded there will be no increase in the maximum credible sustained overvoltage of the low voltage system whether a ground fault is present at the medium voltage system or not. As such, the maximum credible sustained overvoltage numerical value, when the transformer is lightly loaded, can simply be derived from consideration of the nominal low voltage system of 480VAC plus a 10 percent multiplier (480VAC)(1.10) = 528 VAC for an maximum sustained system voltage during both normal full load and light load operation. The high voltage taps of the load center transformer is set such that the maximum allowable voltage at the terminals of the loads and the secondary winding of the load center transformers is not exceeded.

For purposes of defining a value for design of isolation devices a margin of 10 percent will be used yielding (528VAC)(1.10) = 580VAC. This value is developed specifically for use as a bounding design value for isolation devices and, as described above, is conservative beyond the actual design operating conditions of the plant.

This established maximum credible over-voltage value allows for the identification of operating characteristics of the IITA and MI cable hardware used in the IIS to be specified to withstand a peak over-voltage beyond the identified historical maximum over-voltage value. The design requirements for the MI cable and IITA electrical connector hardware require that manufacturing or proof testing be performed to demonstrate compliance with a 600 volts alternating current (VAC) peak voltage CET functional interaction exclusion requirements. This hardware testing requirement satisfies the requirements for testing or analysis of associated circuit interaction with Class 1E circuits in IEEE Std. 384-1981 for over-voltage conditions. To further mitigate the possibility of a transient surge voltage condition in the SPS cabinet's input power supply in excess of the identified maximum over-voltage value that may disable both divisions of the CET signals used by the PAMS, different divisions of safety power are supplied to the IIS SPS cabinets, with the power cables routed in separate shielded conduits.

The applicant also identified the DAS as another non-Class 1E system that can cause a surge or over-voltage faults to propagate to the IIS. The applicant’s analysis found that the maximum credible surge voltage output from the DAS to the DAS CET signal leads that could produce an interaction with the IIS is the same as the IIS IITA and CET cable. The identified maximum credible voltage output from the DAS to the CET signal leads are also equivalent to the electrical connector hardware voltage environmental electromagnetic interference qualification limit requirements in Tier 2, Appendix Section 3D.4.1.2 of the AP1000 DCD. The DCD hardware requirements specifically require that the IIS IITA and associated cables be qualified to meet RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference for Safety-Related Instrumentation and Control Systems,” peak surge voltage pulse levels. This result ensures that if there is a voltage surge from DAS that propagates down through the DAS CET signal leads to the associated SPD cables, there will be no credible, systematic shorting of DAS CET signals to the associated SPD signal leads, which contain the CET signals that are sent to the PMS.

IEEE Std. 603-1991, Clause 5.6.3, requires that equipment in other systems that is in physical proximity to safety system equipment but that is neither an associated circuit nor another Class 1E circuit be physically separated from the safety system equipment to the degree necessary to retain the safety systems' capability to accomplish their safety functions in the event of the failure of nonsafety equipment. Physical separation may be achieved by physical barriers or acceptable separation distance. The separation of Class 1E equipment shall be in accordance with the requirements of IEEE Std. 384-1981.

The NRC issued RG 1.75, "Criteria for Independence of Electrical Safety Systems" to endorse IEEE Std. 603-384 with exceptions. RG 1.75 identifies the underlying separation criteria: (1) physical separation, and (2) electrical isolation are provided to maintain the independence of safety-related circuits and equipment so that the safety functions required during and following any design-basis event can be accomplished. Section 5.6(3) of IEEE Std. 384-1992 provides general criteria for independence between safety-related and nonsafety-related circuits. When minimum separation cannot be met, it allows an analysis of nonsafety-related circuits to demonstrate that the safety-related circuits are not degraded below an acceptable level. If the analysis is successful, the nonsafety-related circuits can remain as nonsafety-related circuits. RG 1.75 clarifies that: (1) nonsafety-related circuits that are not separated from safety-related circuits through the minimum separation or barriers must be treated as "associated circuits," and (2) the cables that are associated because they are powered from a safety-related source serving nonsafety-related loads or share the safety signal must also be treated as associated circuits. This regulatory guide defines "associated circuits" as "nonsafety-related circuits that are not physically separated or not electrically isolated from safety-related circuits by acceptable separation distance, safety class structures, barriers, or isolation devices."

Based on the staff's evaluation of the information presented in WCAP-17226, the staff finds that the associated circuit analysis performed and the ensuing design requirements adequately ensure that the Class 1E CET elements cannot be degraded below an acceptable level by over-voltage or surges from non-Class 1E SPD elements or other connected nonsafety systems. The specific evaluation of portions of this analysis is documented below:

- Within the IITA, the staff finds that placing the active portions of the Class 1E CET elements and the non-Class 1E SPD elements inside individual steel outer sheaths that share a common ground provides adequate electrical isolation between the CET and SPD elements. The staff finds that since the steel outer sheaths share a common ground, it can adequately protect the CET element from a potential overload of the SPD element.

- From the Quickloc flanges to the RDP, the staff finds the MI cable assembly adequately ensures the isolation of Class 1E CET elements from SPD elements through the use of separate individual flexible steel-sheathed cable subassemblies for the CET and SPD signal leads. The individual steel-sheathed cables provide adequate electrical isolation to prevent faults within the SPD signal leads from propagating to the CET signal leads.

  The staff finds adequate the analysis provided to evaluate the maximum credible over-voltage or surge voltage that can propagate backwards from the non-Class 1E power supplies in the SPS cabinets to the SPD input signals. Specifically, the staff finds that the applicant's analysis, which specifies that, for 3-wire low voltage AP1000 systems, the maximum sustained over-voltage will incorporate a 10 percent margin (which is above the 5 percent margin specified in ANSI C84.1-2006, "Electric Power Systems and Equipment - Voltage Ratings (60 Hz)") for a solidly grounded system is consistent with the criteria found in IEEE Std. 141-1993, "IEEE Recommended Practice for Electric Power Distribution for Industrial Plants." IEEE Std. 141-1993, Clause 7.2.5, states there are three levels of conductor insulation for medium-voltage cables that are permitted: 100, 133, and 173 percent. The solidly grounded system permits the use of 100 percent insulation level, which indicates that the maximum sustained over-voltage is at 100 percent of the line-to-ground voltage during a single line to ground fault. Thus, the 10 percent additional margin sufficiently bounds the anticipated maximum sustained over-voltage. Ultimately, the cable insulation selection for a nominal low voltage system of 480 VAC will be rated at a minimum of 600 volts (V), which encompasses the anticipated maximum sustained over-voltage value of 528 VAC. The derivation of the maximum credible over-voltage was incorporated in Revision 2 to WCAP-17226.

- The applicant proposed to specify design requirements for the MI cable and IITA electrical connector hardware to be tested to withstand the identified maximum credible voltage values. This approach is acceptable to the staff in that it meets the associated circuit requirements of IEEE Std. 384-1981 for over-voltage conditions. In addition, assigning each SPS cabinet and its corresponding PAMS division to a different Class 1E power bus, the staff finds this approach ensures any fault on the SPS cabinets' safety input power supplies will only occur on one SPS cabinet and can, therefore, only disable one division of the CET signals used by PAMS. The staff finds this acceptable.

- Furthermore, the staff finds adequate the identified maximum credible overvoltage that can be generated from the DAS that could produce an interaction with IIS. The staff finds that by qualifying the IIS IITA and associated cables in accordance with RG 1.180, the design characteristics ensure that a potential voltage surge from the DAS will not be able to cause systematic shortening of the DAS CET signal to SPD signal leads. Therefore, the staff finds that the PAMS CET coverage is adequately protected against failures of the DAS.

7.5.9 Evaluation Findings and Conclusions

Appendix D.II, "Definitions," to 10 CFR Part 52, defines COL action items (COL license information) as items that identify certain matters that applicants must address in the site-specific portion of the final safety analysis report (FSAR). The applicant selected the site-specific parameters required to meet 10 CFR 50.34(f)(2)(xvii), regarding accident monitoring instrumentation as a COL action item and updated Tier 2 Tables 1.8-1 and 1.8-2, accordingly. Therefore, the staff finds the above requirements have been satisfied via the commitment in the applicant's response dated May 26, 2010. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Regulations in 10 CFR 50.34(f)(2)(v) require the applicant to provide for automatic indication of the bypassed and inoperable status of safety systems. The staff finds that the applicant has sufficiently addressed this requirement in the AP1000 DCD and the supporting TRs, as documented in Section 7.9.2.6. Specifically, the applicant has demonstrated how the indication of bypassed and operable status of safety systems in the MCR conforms to RG 1.47. The staff previously identified this issue as Open Item OI-SRP7.5-ICE-02, and based upon the discussion in Section 6.4 of WCAP-16675, the staff finds the issue has been satisfactorily resolved.

Based upon the staff's review of WCAP-17226-P, the interaction between the IIS and other safety-related systems is deemed acceptable provided the proposed change to WCAP-17226-P to include a discussion of how the maximum credible over-voltage value is derived is included in

Revision 2 of the document. These changes were incorporated in Revision 2 to WCAP-17226-P, which resolves this issue.

Additionally, the staff finds that the accident monitoring instrumentation meets the applicable requirements of GDC 13 and GDC 19, “Control Room.”

7.6 Interlock Systems Important to Safety

7.6.5 Evaluation Findings and Conclusions

Section 7.3.1.4.1 of this report describes the applicant’s commitment to add an ADS valve blocking circuit to prevent the spurious actuation of any of the valve paths in any of the stages for the ADS. As this circuit is described as a blocking circuit rather than an interlock (in which a block may be cleared or overridden but an interlock must not), the circuit’s operation is not captured in Section 7.6 of this report. As such, the changes in Section 7.6 of the AP1000 DCD did not affect any conclusions regarding regulatory compliance in NUREG-1793. Therefore, the staff finds the applicant continues to meet the requirements identified in NUREG-1793 for Section 7.6.

7.7 Control and Instrumentation Systems

7.7.1 System Description

7.7.1.1 Reactor Power Control System

The applicant revised the second paragraph of Section 7.7.1.1.1, “Power Control,” to remove the term “lead/lag compensated.” The applicant revised the AP1000 design to apply the lead/lag compensation after the high auctioneer, versus before it. The applicant claims that this design does not require compensation to be factored into signal quality check acceptance criteria in the auctioneer. The applicant described this justification in APP-GW-GLR-080, Revision 0, “Mark-up of AP1000 Design Control Document Chapter 7.” The staff’s review confirms that this design change does not affect any staff conclusions in NUREG-1793.

The applicant also revised the description of the T_{ave} reactor control band for the various plant modes of control (i.e., load follow, load regulation, base load). The applicant claims that there is no advantage to increasing the deadband during load follow operations. The applicant further states that doing so would erode margins to reactor trip setpoints. The applicant also stated this justification in APP-GW-GLR-080, Revision 0. The staff’s review confirms that this change does not affect any staff conclusions in NUREG-1793.

The applicant removed the “time weighted average” or “smoothing” compensation to the nuclear flux and the axial offset signals while the plant is in a load regulation mode of control. The applicant states that this current transient specification does not require complex “time weighted average” nuclear flux and axial offset signal compensation on the inputs to the axial offset control band calculation and that simple lag compensation is adequate. The applicant also stated this justification in APP-GW-GLR-080, Revision 0. The staff’s review confirms that this change does not affect any staff conclusions in NUREG-1793.

7.7.1.2 Rod Control System

The applicant revised the interlock and “low” and “low-low” alarms associated with control rod insertion limits. The revision will move the automatic activation of the rod insertion interlock from the “low” rod insertion alarm setpoint to the “low-low” rod insertion alarm setpoint. In moving the activation of the rod insertion interlock from the “low” to the “low-low” rod insertion alarm, rod insertion will be prevented, at the “low” alarm setpoint, by following appropriate plant operating procedures and will be prevented at the “low-low” setpoint by automatically actuating the rod insertion interlock and terminating automatic AO bank insertion (or withdrawal to prevent further M bank insertion).

By moving the rod insertion interlock to the “low-low” alarm setpoint, continued rod insertion (and thus, a continued reduction in control rod shutdown margin) is automatically terminated by plant controls versus “appropriate plant operating procedures.”

Furthermore, removing the interlock from the “low” alarm setpoint does not reduce any plant protection function or increase the risk of reducing protection against a reduction in control rod shutdown margin caused by the margin built into the difference between the “low” and the “low-low” rod insertion setpoints. Therefore, the staff’s review confirms that this design change does not affect any staff conclusions in NUREG-1793, including Supplement 1.

7.7.1.3 Pressurizer Pressure Control

The applicant changed the description of the pressurizer variable heating control by stating that it is not sensitive to the rate of change in pressure and that it will respond in the same manner to small, fast, or slow small changes in pressure. The staff’s review confirms this change does not affect any staff conclusions in NUREG-1793.

7.7.1.5 Feedwater Control

In Section 7.7.8.1 of the AP1000 DCD, the low-range feedwater flow measurement is no longer used in the low-power mode, and it is not used in the integration (reset) action of the low-power mode feedwater flow controller. This means that only feedwater temperature (low-power mode) and steam flow (high-power mode) are used to tune the integrator setpoints.

In both Sections 7.7.8.1 and 7.7.8.2, the control of the lift on the main and startup feedwater control valves is no longer determined by the ΔP available across the feedwater control valve, and the flow coefficient (C_v) characteristic of the valve. Therefore, in high-power control mode, the feedwater flow is regulated in response to changes in steam flow and proportional plus integral (PI)-compensated steam generator narrow-range water level deviation from setpoint. In the low-power control mode, the feedwater flow is regulated in response to changes in steam generator wide-range water level and PI-compensated steam generator narrow-range water level deviation from setpoint.

The startup feedwater control subsystem regulates the flow of feedwater in a manner similar to the way the (main) feedwater is controlled in the low-power control mode. Feedwater flow is regulated in response to changes in the steam generator wide-range water level and PI-compensated steam generator narrow-range water level deviation from setpoint. The staff’s review confirms that this design change does not affect any staff conclusions in NUREG-1793.

7.7.2 Diverse Actuation System

Section 7.8 includes a detailed evaluation regarding the DAS design changes.

7.7.3 Signal Selector Algorithms

The applicant has not demonstrated what specific actions are taken by the signal selector algorithms (SSAs) in the event that one of the multidivisional or multichannel inputs is deemed faulty or of "bad" quality. The staff requires all outputs of the device, whether they are in the form of control, alarm, interlock, or indications, to be identified and addressed. The staff issued this request to the applicant as RAI-SRP7.7-ICE-01. The applicant responded to the RAI in a letter, dated July 7, 2008. However, the staff found the response inadequate. The applicant submitted Revision 1 to address this issue on May 6, 2009, and subsequently the applicant submitted Revision 2 of RAI-SRP7.7-ICE01. In the Revision 2 response, submitted on January 27, 2010, the applicant states it will update the forthcoming revision of the AP1000 DCD so that it is clear that the SSAs are executed in the PLS and additionally that PMS and DAS performance are independent of the SSA. The staff previously identified this as Open Item OI-SRP7.7-ICE-01. Revision 2 of RAI-SRP7.7-ICE-01 adequately addresses this issue. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

7.7.4 Evaluation Findings and Conclusions

The staff finds the conclusions described in NUREG-1793 still valid, based on the staff's review of the changes proposed in the AP1000 DCD. Specifically, the staff required additional information on how the SSAs would affect the PMS and the DAS. Information on such impacts could affect the degree of independence between the control system and the protection system, as required in GDC 24. Also, such impacts could affect the degree of diversity and quality of the DAS as required in GDC 22. The response discussed in Section 7.7.3 above, adequately addressed the issue. The staff previously identified this as Open Item OI-SRP7.7-ICE-01.

Section 7.8 includes further evaluation of the DAS.

7.8 Diverse Instrumentation and Control Systems

7.8.1 System Description

The I&C systems reviewed in this section include the diverse I&C systems and equipment that provide a diverse backup to the safety-related PMS and the defense against postulated CCFs in the PMS and the nonsafety-related PLS concurrent with postulated transients.

The review ensured that the applicant designed and installed the anticipated transient without scram mitigation systems and equipment in accordance with the requirements of 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," and 10 CFR Part 50, Appendix A, GDC 22.

The DAS in the AP1000 DCD is a nonsafety related I&C system and provides a diverse and independent method for tripping the reactor and performing several ESFs in order to meet the requirements of 10 CFR 50.62 and 10 CFR Part 50, Appendix A, GDC 22. In addition, a set of dedicated and independent displays of selected plant indications and manual controls is provided in the MCR to address the criteria in BTP 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems."

The scope of the safety evaluation in the following section is limited to the changes that have been made to the approved DAS system since the AP1000 DCD, Revision 15, was certified.

7.8.2 Diverse Actuation System Assessment

The findings in NUREG-1793, Supplement 0, related to DAS functionality are based upon WCAP-15775, "AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report," (APP-GW-J1R-004), Revision 0, issued April 2002. The applicant submitted Revision 2 of its D3 assessment on April 3, 2003. The applicant revised its D3 assessment based on the response to the NRC's RAI 420.013, Revision 1, in a letter dated February 21, 2003. The applicant referred to the Revision 2 of the D3 assessment in the certified AP1000 DCD, Revision 15. In the certified AP1000 design the hardware of the DAS is microprocessor based and its operating systems or programming languages are different from those used in the PMS.

In the original NUREG-1793, Section 7.1.6, the staff concluded that "the proposed design satisfies the Commission's position on I&C system diversity." The NRC based its conclusion on the proposed DAS design, as stated in Revision 15 of the applicant's DCD, which included the ITAAC listed in AP1000 DCD Tier 1, Section 2.5.1.

In the second paragraph of Section 7.7.2 of NUREG-1793, the staff concluded that based on the design commitments of Revision 15 of the AP1000 DCD:

> The DAS automatic actuation is accomplished by a microprocessor-based system. Diversity from the PMS is achieved by using a different architecture, different hardware implementation, and different software. Software diversity is achieved by running different operating systems and programming in a different language.

The applicant submitted TR-97, Revision 1, in which it changed the DAS design commitments and DAS ITAAC in Revision 15 of the AP1000 DCD to allow non-microprocessor-based implementations, added the DAS instrumentation cabinet, added an electrical penetration to the containment, and relocated a portion of the DAS to another area of the plant.

The following evaluation focuses on the modification to allow non-microprocessor-based implementations of the DAS and the addition of the DAS instrumentation cabinet. The relocation of DAS equipment within the plant has no impact upon the I&C review. Additionally, the installation of another electrical penetration to the containment is beyond the scope of the Chapter 7 review.

For the purposes of the Chapter 7 I&C review, the following areas of the DCD are affected:

Tier 1
Section 2.5.1
Table 2.5.1-4
Table 2.5.1-5
Table 3.7-1
Tier 2
Section 7.7.1.11
Figure 7.2-1
Section 9A.3.1.3.1.1
Table 14.3-3
Table 14.3-6
Table 17.4-1
Table 19.59-18

The applicant discussed the change to the I&C technology utilized for the DAS throughout Tier 1 and Tier 2 in Revision 16 of the AP1000 DCD by substituting language, as appropriate, where it used the terms “microprocessor” and “software” to describe DAS technology. The applicant replaced the term “microprocessor” with “microprocessor or special purpose logic processor,” and the term “software” with “any software.” The addition of the DAS instrument cabinet, DAS-JD-004 will contain the 4-20 mAdc loop transmitters and power supplies necessary to complete the DAS instrumentation requirements. The DAS-JD-004 cabinet mounts next to the DAS squib valve control cabinet, DAS-JD-003. This is acceptable. The applicant revised Section 7.7.1.11 of the AP1000 DCD Tier 2, Revision 16, by replacing the terms “microprocessor-based” and “microprocessor” with “logic”, “software” with “any software” from several design change descriptions in APP-GW-GLR-080, Revision 0, which describes all changes to the AP1000 I&C systems in the certified AP1000 DCD.

This proposed change to use non-microprocessor-based technology is intended to increase reliability and was found to be acceptable. However, a discussion concerning the diversity between the PMS and the DAS is provided later in this section.

In Revision 17 of the AP1000 DCD, the applicant removed DAS DAC from Tier 1, Section 2.5.1, “Diverse Actuation System (DAS) Design Description,” Items 4a and 4b, as well as from Tier 1, Table 2.5.1-4, ITAAC Items 4a and 4b. Items 4a and 4b are the design requirements and system definition phases of the DAS hardware and software design process. The applicant removed portions of the DAC, and also provided the corresponding design information in WCAP-17184-P, Revision 2, which addresses those two phases. However, the applicant failed to provide a description in Chapter 7 of the AP1000 DCD, Tier 2, regarding completion of those two phases found in the AP1000 DCD Tier 1. 10 CFR 52.47(a)(2) requires standard DC applications to provide a level of design information sufficient to enable the Commission to reach a final conclusion on all safety issues associated with the design before the certification is granted. In the proposed amendment to the AP1000 DCD, Tier 1, ITAAC Items 4a and 4b are removed based on design work accomplished. In Chapter 7 of the AP1000 DCD, Tier 2, the applicant should provide a summary and justification for why the ITAAC found in Items 4a and 4b of AP1000 DCD, Tier 1, Table 2.5.1-4, can be removed. The staff issued RAI-SRP7.8-DAS-12, requesting the applicant describe the completeness of the above two phases in Chapter 7 of the AP1000 DCD, Tier 2. In response to this RAI, the applicant provided a detailed description about the two completed life cycle phases as a markup for AP1000 DCD Tier 2, Section 7.7.1.11, which is found acceptable. In addition, the applicant added an ITAAC item to Table 2.5.1-4 of the AP1000 DCD Tier 1 to address the ITAAC for DAS manual actuations. The staff finds the above changes to the AP1000 DCD Tier 1 and Tier 2 acceptable and finds the response to RAI-SRP7.8-DAS-12 acceptable. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In the initial submittal of WCAP-17184-P, Revision 0, the staff found that the applicant failed to address the DAS setpoint methodology. GDC 13 requires, among other things, that appropriate controls be provided to maintain variables and systems within prescribed operating ranges. The guidance in BTP 7-19, Section B, Item 3, Positions 1 and 2, acceptance criteria address confirmation that an anticipated operational occurrence (AOO) and postulated accidents are mitigated in the presence of CCF.

In WCAP-17184-P, Revision 0, the applicant did not identify how the DAS actuation setpoints and timing delays would be established. DAS must be able to perform its functions to ensure the potential release of radioactive material for postulated accidents and AOOs fall within acceptable limits in the event of a software CCF of the safety-related protection system. The staff issued RAI-SRP7.8-DAS-02, requesting that the applicant describe the DAS setpoint methodology. To address this RAI, the applicant provided WCAP-17184-P, Revision 1, to add the DAS setpoint methodology description as a new appendix to this TR. After reviewing the DAS setpoint methodology description, the staff found that the DAS allowances for automatic actuation signals, “Containment Temperature High” and “Pressurizer Water Level Low” are outside of the typical ratio of 1.15σ/2σ. The staff issued RAI-SRP7.8-DAS-10, requesting that the applicant provide the design basis to support the deviation. In response to this RAI, the applicant proposed a revision to WCAP-17184-P in which it described why the DAS channel statistical allowances for “Containment Temperature High” and “Pressurizer Water Level Low” do not conform to the typical ratio of 1.15σ/2σ when comparing a 75 percent probability/75 percent confidence level to 95 percent probability/95 percent confidence level for determination of the random and independent terms of the square-root-sum-of-the-squares calculation. Additionally, the applicant also provided a justification for the use of a 75 percent probability/75 percent confidence level. After reviewing the revised report, the staff found that the applicant adequately addressed the DAS setpoint methodology and found it acceptable. Therefore, the staff finds the responses to RAI-SRP7.8-DAS-02 and RAI-SRP7.8-DAS-10 acceptable. These changes were incorporated into Revision 2 to WCAP-17184-P.

In the proposed changes to the AP1000 DCD, the applicant changed the microprocessor-based implementation of the DAS to be a special purpose logic processor-based system. This special purpose logic processor-based DAS is further described as an FPGA digital platform-based system in WCAP-17184-P. The applicant also made changes to use the FPGA technology for the CIM in the safety-related PMS in WCAP-17179-P and WCAP-16675-P. According to the above reports, the CIM and DAS systems will be designed and manufactured by the same company at a common design and manufacturing facility. 10 CFR Part 50, Appendix A, GDC 22, requires, among other things, that design techniques such as functional diversity or diversity in component design and principles of operation shall be used to the extent practical to prevent loss of the protection function. BTP 7-19 provides guidance for evaluating an applicant’s D3 assessment to ensure conformance with the NRC position on D3 for digital I&C systems. NUREG/CR-6303, “Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems” provides diversity analysis methods and strategies to demonstrate that adequate and sufficient diversity should be included within the design. The staff found that the applicant had not provided design descriptions capable of demonstrating adequate and sufficient diversity between the DAS and CIM systems in accordance with the guidance listed above. Hence, the staff issued RAI-SRP7.8-DAS-04, requesting that the applicant describe in detail how the DAS equipment (i.e., hardware, software) would be diverse from the safety-related CIM in PMS. The staff also issued another RAI-SRP7.8-DAS-05 requiring the applicant to identify the criteria, practices, and processes that will ensure adequate diversity in the development of the CIM and the DAS at the common design and manufacturing facility, including the diversity with respect to human, software, and equipment diversity.

In response to the above RAIs, the applicant states, in part, that diversity is a principle in instrumentation of sensing different variables, using different technology, using different logic or algorithm, or using different actuation means to provide different ways of responding to postulated plant conditions. The applicant also revised WCAP-15775 to Revision 4 to address the specific requirement of diversity between CIM and DAS. The applicant demonstrated in Section 2.11 of WCAP-17179-P, Revision 2, and Section 9 of WCAP-17184-P, Revision 2, how the requirements of diversity are met between CIM and DAS. For example, for the human diversity, the applicant states in the two TRs that different designers are used for the CIM and DAS designs. In addition, the different design teams and different test teams will be used to test the CIM and DAS designs. In order to achieve the software diversity between the DAS and PMS (i.e., CIM), the applicant will use different algorithms, logic, program architecture, executable operating system and executable software/logic. The staff concludes that the applicant has provided sufficient information demonstrating conformance with regulatory policies and criteria concerning diversity. The AP1000 DCD Tier 1 and Tier 2 will also be updated accordingly to address the software diversity. Therefore, the staff finds the responses to RAI-SRP7.8-DAS-04 and RAI-SRP7.8-DAS-05 acceptable.

The staff found inconsistencies between Chapter 7 of the AP1000 DCD, Tier 2, Revision 17 and WCAP-17184-P, Revision 1, regarding DAS manual actuations. 10 CFR 52.47(a)(2) requires that an application must include a sufficient description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justificationupon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Therefore, the staff issued RAI-SRP7.8-DAS-06 to request the applicant provide correct and unambiguous design descriptions for the list of DAS manual actuations. The applicant clarified that the list in the AP1000 DCD is all-inclusive of any manual actuation performed by DAS. The tripping of the RCPs is only done in conjunction with the CMT actuation and, therefore, not separately listed in the DCD. The list in Section 2.3 of WCAP-17184-P, Revision 1, is the list of manual actuations that are cited in the probabilistic risk assessment (PRA) for enabling AP1000 to meet its large release frequency (LRF) goal for beyond design basis events. Therefore, this list would be different from the list in the DCD because not all of the DAS manual actuations are needed to meet the LRF goal for beyond design basis events. Table B-1 in WCAP-17184-P is also not intended to be a complete list of all of the manual actuations. This table is modified to only include those manual actuations that do not have automatic DAS actuations. To incorporate the above clarification, the applicant proposed a revision to WCAP-17184-P, which the staff found acceptable. Therefore, the staff finds the response to RAI-SRP7.8-DAS-06 acceptable. These changes were incorporated into Revision 2 to WCAP-17184-P, which resolves this issue.

The staff found Appendix B, Table B-1, in WCAP-17184-P, Revision 2, includes the manual actuation of the hydrogen control system or igniters, but it is not credited in Section 2.3 of WCAP-17184-P for the DAS manual operator action and/or a DAS automatic function. The applicant did not provide a technical basis for this non-credited DAS manual operator action. BTP 7-19 states where operator action is cited as the diverse means for response to an event, the applicant should demonstrate that adequate information (indication) and sufficient time is available for operator action. The staff's review of Section 10.2.1.1 in WCAP-17184-P, Revision 2, determined that the stated design descriptions do not provide an explanation of how the manual action is used in the DAS design beyond listing the manual actions. From the PRA evaluation the staff found that there is a 19-minute window for accomplishing this action. The applicant failed to provide a clear technical basis that will permit sufficient understanding of credited DAS manual actuations and their conformance with the applicable regulatory requirements. The staff issued RAI-SRP7.8-DAS-07 to request the applicant provide the technical basis for the hydrogen igniter manual action as a non-credited DAS function.

In response, the applicant states, in part, that for this manual actuation of the hydrogen control system or igniters, PRA analysis techniques show acceptable results even if the act of manually actuating the hydrogen igniters is not accomplished (operator fails to act with 100 percent certainty). For this reason, manual actuation of the hydrogen igniters is not a credited manual operator action nor is it required (or credited) for hydrogen igniters to operate automatically. The 19-minute window as described in the PRA analysis is for a beyond design basis event. The PRA analysis was used to provide insights into this particular scenario. Hydrogen igniters were added to the AP1000 design even though they are not credited in the design basis. In response to this RAI, the applicant revised WCAP-17184-P to provide the reasoning behind installation of hydrogen igniters. After evaluating the response, the staff found the applicant's response to this RAI acceptable. Therefore, the staff finds the response to RAI-SRP7.8-DAS-07 acceptable. These changes were incorporated into Revision 2 to WCAP-17184P, which resolves this issue.

In Appendix B, Table B-1, in WCAP-17184-P, Revision 1, the staff found that there is a 20-minute window for accomplishing the manual actuation of the ADS. However, after reviewing the above TR, the staff found that the stated design descriptions do not provide an explanation of how the ADS manual actions are used in the DAS design beyond listing them. According to BTP 7-19, the applicant should demonstrate that adequate information (indication) and sufficient time is available for manual operator actions. The staff issued RAI-SRP7.8-DAS-08 to request the applicant provide a clear technical basis description in the TR that permits sufficient understanding of ADS manual operator action as credited DAS manual actuations and the basis for why the 20-minute window is sufficient for completing the ADS manual actuation. In response to this RAI, the applicant revised WCAP-17184-P to provide clarification for this manual actuation of ADS.

The DAS credits manual actions to depressurize the RCS during AOOs or postulated accidents following a software CCF in the protection system. BTP HICB-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems," Revision 4, states that where operator action is cited as the diverse means for response to an event, the applicant should demonstrate that adequate information (indication) and sufficient time is available for operator action. The staff reviews acceptability of these manual actions using guidance in DI&C-ISG-05, "Highly-integrated Control Rooms – Human Factors Issues" (ISG-5).

Manual actions that can be initiated from DAS are listed in the following places:

- AP1000 DCD, Revision 19, Tier 2, Section 7.7.1.11
- WCAP-17184-P, Revision 1, Section 2.3
- WCAP-17184-P, Revision 1, Appendix B, Table B-1

The manual actions listed in WCAP-17184-P, Section 2.3, were not consistent with the other two lists. The staff initiated RAI-SRP7.8-DAS-06 to resolve the differences. In its response of June 22, 2010, the applicant explained that the DC described all manual actions that can be performed by DAS. WCAP-17184-P, Section 2.3, described the manual actions that are cited in the PRA for enabling AP1000 to meet its LRF goal for beyond design basis events. The lists are different because not all of the DAS manual actions are needed to meet the LRF goal for beyond design basis events. Appendix B, Table B-1, provided a list matching the DCD. The applicant will modify Table B-1 to include only those manual actions that do not have corresponding automatic DAS actuations. For these manual actions, the applicant provided specific information describing whether the manual action is credited as part of the DAS response to a CCF or whether the manual action is part of the defense-in-depth strategy for severe accident management, thus providing a clearer communication of why the manual actions are included in the DAS design. These changes were provided in Revision 2 of WCAP-17184-P. The staff found the changes acceptable. These changes were incorporated in a Revision 2 to WCAP-17184-P, which resolves this issue.

Appendix B of WCAP-17184-P now lists the following 4 manual actions:

Manual Action 1: Manual Initiation of IRWST Recirculation/IRWST Drain for In Vessel Retention Support

The DAS provides this capability as part of the defense-in-depth strategy for severe accident management. It is an action needed to address AOOs or postulated accidents following CCFs in the protection systems. Therefore, the regulatory guidance in BTP 7-19 and ISG-05 apply.

Manual Action 2: Manual Initiation of the Hydrogen Control System

The DAS cabinet presents a convenient and reliable location for the manual hydrogen control system switches because it is in the MCR and has a diverse battery-backed power supply. It is not an action needed to address AOOs or postulated accidents following CCFs in the protection systems. Therefore, the regulatory guidance in BTP 7-19 and ISG-05 does not apply.

Manual Action 3: Manual Depressurization of the RCS

This action is credited in the DAS response to a CCF. In summary, ISG-05 states that an analysis must be completed that demonstrates:

- The time available to perform the required manual actions is greater than the time required for the operator(s) to perform the actions.

- The operator(s) can perform the actions correctly and reliably in the time available

No information was provided to the staff to explain how this guidance was addressed. The staff initiated RAI-SRP7.8-DAS-8 requesting this information. In its response, dated June 22, 2010, the applicant committed to include a new ITAAC in Tier 1 Chapter 2, Section 2.5.1, “Diverse Actuation System,” Table 2.5.1-4. This commitment was provided in the response to RAI-SRP7.8-DAS-12 and reads as follows:

Table 2.5.1-4. Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The DAS manual actuation of ADS, IRWST injection, and containment recirculation can be executed correctly and reliably.	An evaluation to confirm that the operator actions can be performed within the specified times.	b) A report exists and concludes that DAS manual operator action verification was conducted.

The staff finds the proposed ITAAC to be an acceptable way to develop the information needed to address the regulatory guidance in ISG-05, provided the commitment is also included as a separate “Design Description” in the text-based portion of Section 2.5.1. In a subsequent revision of the AP1000 DCD, the applicant added Design Description 5 and Design Commitment 5 to Tier 1 Section 2.5.1 and Table 2.5.1-4 respectively, which resolves this issue.

The staff notes that APP-GW-GL-011, “AP1000 Identification of Critical Human Actions and Risk Important Tasks,” Revision 0, Tables 3-1 and 3-2 identify risk-important human actions. Table 3-2, Basic Event ID: AND-MAN01 identifies failure to actuate the ADS for RCS depressurization as recovery from failure of automatic actuation or for manual ADS actuation as a risk-important human action. As such this manual action is already included in the HFE program described in Chapter 18 of this report. In summary, that program includes all risk important human actions as priority items in the task analysis, the human-system interface (HSI) design, the HFE design V&V, and human performance monitoring elements. By being included in the HFE program the regulatory guidance provided in ISG-05 is either met or exceeded.

Manual Action 4: Manual Initiation of IRWST Gravity Injection.

The DAS provides this capability as part of the defense-in-depth strategy for severe accident management. It is an action needed to address AOOs or postulated accidents following CCFs in the protection systems. Therefore, the regulatory guidance in BTP 7-19 and ISG-05 applies.

Operator manual actions credited within the DAS design have not yet been evaluated to verify they are viable. The applicant provided an acceptable DAS ITAAC in Section 2.5.1 of Tier 1, Chapter 2 information that will track completion of this evaluation. A similar evaluation is required within the HFE program described in DCD Chapter 18. Together or independently these commitments will ensure the regulatory guidance in BTP 7-19 and ISG-05 is implemented. This, in turn, will provide reasonable assurance that the operator manual actions to depressurize the RCS are an effective element of the DAS design for coping with AOOs or postulated accidents following CCFs in the protection systems.

Additionally, by the applicant adding a new ITAAC item to Table 2.5.1-4 of the AP1000 DCD Tier 1 to address the design commitment and ITAAC for DAS manual actuations, the staff found the applicant’s response to this RAI acceptable. Therefore, the staff considers RAI-SRP7.8-DAS-08 closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff found in Appendix B, Table B-1, in WCAP-17184-P, Revision 1 that the applicant includes IRWST gravity injection as an DAS manual actuation. From the PRA evaluation in the table, there is a 20-minute window for accomplishing this manual action. However, the applicant failed to provide the technical basis for the sufficiency of this manual operator action. The applicant should demonstrate that adequate information and sufficient time is available for manual operator actions based on the guidance in BTP 7-19. Therefore, the staff issued RAI-SRP7.8-DAS-09 to request the applicant identify a clear technical basis description that permits sufficient understanding of this credited DAS manual actuation and its basis for why the 20-minute window is sufficient for this DAS manual action. In response to this RAI, the applicant states, in part, that for this particular scenario, PRA analysis techniques show acceptable results even if the act of manually actuating IRWST gravity injection is not accomplished (operator fails to act with 100 percent certainty). For this reason, manual actuation of IRWST gravity injection is not a credited manual operator action nor is it required (or credited) for automatic operation. The 20-minute window as described in the PRA analysis is for a beyond design basis event. The PRA analysis was used to provide insights into this particular scenario. The capability to manually actuate IRWST gravity injection from DAS was added to the AP1000 design out of caution, even though it is not credited in the design basis. The applicant proposed to revise WCAP-17184-P to Revision 2 to provide further clarification of the reasoning behind the manual initiation of IRWST gravity injection. The applicant committed to add a new ITAAC item to Table 2.5.1-4 of AP1000 DCD Tier 1 to address the design commitment and ITAAC for DAS manual actuations, which include the manual actuation of the IRWST gravity injection. The staff found that the applicant’s response and changes related to this RAI are acceptable. The staff finds the response to RAI-SRP7.8-DAS-09 acceptable. These changes were incorporated into Revision 2 of WCAP-17184-P, which resolves this issue.

In WCAP-17184-P, Revision 1, which the applicant submitted to justify the removal of ITAAC Items 4a and 4b from AP1000 DCD Tier 1, Table 2.5.1-4, the staff found that the applicant addressed the requirements of cyber security. The requirements of cyber security were also addressed in WCAP-17184-P, Revision 1. However, the staff’s position related to cyber security issues is that cyber security is addressed by 10 CFR 73.54, “Protection of digital computer and communication systems and networks,” and is not a 10 CFR Part 50 review item. As such, the reference to cyber security in the above TRs should be modified and/or replaced with a docketed TR describing the SDOE in which the applicant chooses to develop its software-based and programmable technology based DAS and CIM, paying particular attention to IEEE Std. 603-1991, Clauses 5.3, 5.6.3, and 5.9. The staff issued RAI-SRP7.8-DAS-11, which requested that the applicant remove the discussion of cyber security from the two TRs. In response to this RAI, the applicant deleted the mention of cyber security requirements in the revised WCAP-17184-P and WCAP-17179-P. The applicant also made corresponding changes to AP1000 DCD Tier 2, Table 1.6-1 and Section 7.1 to address the requirements of this RAI. The staff finds the response to RAI-SRP7.8-DAS-11 acceptable. In addition, the applicant submitted APP-GW-J0R-012, Revision 1, to address the secure development and operational environment for the AP1000 PMS. Section 7.9 provides the evaluation of this new TR.

The staff found that the following issues in WCAP-17184-P, Revision 1, need to be addressed by the applicant:

- Section 6.1.2.2 references the Wolf Creek license amendment request regarding self-test features of the DAS. However, the Wolf Creek license amendment request is not part of the AP1000 DCD licensing basis. Therefore, self-test features should be identified in the TR for the AP1000 DAS.

- The statement in Section 8.1 currently states that the “two-out-of-two logic … lends itself to reliability.” However, this configuration is less reliable than a single train configuration, although the PRA-based DAS design and two-out-of-two (2oo2) actuation logic were approved in the certified AP1000 DCD, Revision 15, and the applicant has not made changes to the DAS logic in the applicant’s DC application. Hence, the above statement is not accurate and should be modified to reflect the approved DAS design feature.

- Section 8.1 states that “The use of FPGAs results in a hardware-based design that is not subject to software CCFs. The only software involved in the process is that used to burn-in the required logic design into the FPGA.” These two statements are inaccurate and should be removed because the FPGA-based systems are developed with software tools and can have programming errors similar to microprocessor-based digital systems, although FPGA-based systems do not run any system or application software during operation. Therefore, evaluation for CCFs in the FPGA development process and software programming shall be conducted. A high quality and well documented life-cycle design process should be provided according to BTP 7-14, “Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems.”

During the review of WCAP-17184-P, Revision 1, the staff identified the above issues and issued RAI-SRP7.8-DAS-13 to request that the applicant correct those statements for technical and regulatory accuracy. In response to this RAI, the applicant proposed to modify WCAP-17184-P, Revision 2, to make the necessary corrections. After reviewing the draft revised TR, the staff found the applicant’s response acceptable. The staff finds the response to RAI-SRP7.8-DAS-13 acceptable. These changes were incorporated into Revision 2 of WCAP-17184-P.

The staff finds the DAS is properly credited for providing a diverse backup to the safety-related protection system; however, the ATWS mitigation systems actuation circuitry (AMSAC) system should not have been credited with providing diverse protection upon a postulated CCF of the safety-related PMS. During the evaluation of the changes made to the certified AP1000 DCD, Revision 15 for the DAS, the staff found that the applicant provided ambiguous descriptions for the DAS circuitry and the AMSAC. 10 CFR 52.47(a)(2) requires, in part, a description of the SSCs to be of sufficient detail to permit understanding of the systems designs. Section 15.8 of the AP1000 DCD, states that the DAS provides AMSAC functions. It also states that for Westinghouse plants the ATWS rule (10 CFR 50.62) requires the installation of AMSAC, which consists of circuitry separate from the reactor protection system to trip the turbine and initiate decay heat removal. The applicant failed to provide a description of the AMSAC or the relation between the DAS and the AMSAC system in Section 7.7 of the AP1000 DCD Tier 2. The applicant also failed to clearly describe if the DAS circuitry and the AMSAC system circuitry are the same system or if they are separate systems. The staff issued RAI-SRP7.8-DAS-01 to request the applicant clarify the design descriptions for DAS and AMSAC. In response, the applicant states that for Westinghouse plants the ATWS rule requires the installation of equipment that is diverse from the reactor protection system to automatically trip the turbine and initiate decay heat removal. This equipment must be designed to perform its function in a reliable manner and be independent from sensor output to final actuation device from the existing reactor protection system. The AP1000 is designed with a DAS, which provides the functions required by the ATWS rule. In response to this RAI, the applicant also provided a markup for Section 15.8 of the AP1000 DCD to clarify the description of DAS and AMSAC. The staff found the response to RAI-SRP7.8-DAS-01 acceptable. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

When evaluating the changes to the certified AP1000 DCD, Revision 15, the staff issued RAI-SRP-7.8-DAS-03 requesting the applicant identify design descriptions that demonstrate how the 2oo2 DAS actuation logic would meet the applicable regulatory criteria. 10 CFR 50.62 requires ATWS mitigation equipment to perform its functions in a reliable manner. The guidance of BTP 7-19, Point 3, on D3 states that the diverse or different function may be performed by a nonsafety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. The applicant stated in Revision 17 of AP1000 DCD, Tier 2, Section 16.3-2 that "when a required channel is unavailable, the automatic DAS function is unavailable."

In its response, the applicant explained that there are two actuation logic modes, automatic and manual. The automatic actuation logic mode functions to logically combine the automatic signals from the two redundant automatic subsystems in a 2oo2 basis. The combined signal operates a power switch with an output drive capability that is compatible, in voltage and current capacity, with the requirements of the final actuation devices. The 2oo2 logic is implemented by connecting the outputs in series. The manual actuation mode operates in parallel to independently actuate the final devices. Actuation signals are output to the loads in the form of normally de-energized, energize-to-actuate signals. The normally de-energized output state, along with the dual, 2oo2 redundancy reduces the probability of inadvertent actuation. The staff found the applicant's response to RAI-SRP-7.8-DAS-03 acceptable.

The 2oo2 DAS actuation logic was included in the approved DAS design in the certified AP1000 DCD, Revision 15. Specifically, 10 CFR 50.62 requires that ATWS mitigation equipment be designed to perform its function in a reliable manner. As described in Section 15.8.3 of the AP1000 DCD, the AP1000 is equipped with a DAS, which provides the functions required by the ATWS rule (10 CFR 50.62). The ATWS core damage frequency for the AP1000 is below the SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," goal of 10^{-5} per reactor year. In NUREG-1793, the staff reviewed and approved the AP1000's basis for meeting the ATWS core damage frequency goal.

Reliability of digital systems can be achieved through various means including redundancy, fault detection and management, quality of design, and use of reliable components. Reliability can be defined as the likelihood that a given component or system will be properly functioning when needed, as measured over a given period of time. Reliability, in itself, does not account for any repair actions that may take place. Availability can be defined as the percentage of time that a given system will be functioning as required. In other words, availability is the probability that a system is not failed or undergoing a repair action when it needs to be used.

The AP1000 DAS design addresses reliability from a design/component approach and by fault detection and management. From a design/component reliability approach, Section 8.1 of WCAP-17184-P states that a FMEA, mean-time-between-failure analysis, and a reliability block diagram analysis will be performed on the DAS at the component level. Since the DAS detailed design is not complete at the DC stage, nor required to be complete in accordance with 10 CFR 52.47, those analyses were not part of the staff's review. However, sufficient criteria in the AP1000 DCD are available to guide the detailed design analysis, such as the use of MIL-HDBK-217F for component failures and hardware reliability analysis. From a fault detection/management approach, Section 6.1.2.2.1 of WCAP-17184-P states that the DAS will include self-diagnostic features to identify failures of the processor and supporting circuitry. The self-diagnostic features provide real-time indication to operators of a DAS failure, limiting the fault exposure time and improving DAS availability.

As part of the determination for meeting the ATWS core damage frequency goal, the AP1000 PRA assumed an availability goal for DAS, as described in Section 8.2 of WCAP-17184-P. As committed in WCAP-17184-P, the detailed reliability analysis performed on the DAS would be consistent with the availability goal. Specifically, the reliability analysis will determine an expected failure rate based on hardware failures. Both the failure rate and expected repair time will be calculated and compared to the availability goal for consistency. By utilizing self-diagnostic features, the operators are given real-time indication of a DAS failure, which allows maintenance to be performed in a timely manner. By using the self-diagnostic features, the fault exposure time is reduced on the DAS, thus improving DAS availability as it relates to latent, undetected faults.

Given the commitments in WCAP-17814-P regarding the reliability analysis and self-diagnostics, the staff finds that the DAS will operate reliably. DAS may be taken out of service for maintenance, or be subjected to a failure, but would meet the committed availability target, which is part of the overall basis for meeting the ATWS core damage frequency goal. NRC regulations such as 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” would provide verification that the availability goal is being achieved while the plant is in operation.

As part of the AP1000 design changes, the applicant proposed the addition of a new DAS high hot leg temperature reactor and turbine trip. The reason for the design change is that in the original DAS design, which is modeled in PRA, a reactor trip and turbine trip should occur for ATWS sequences with main feedwater available. Because feedwater is still available, the DAS low steam generator water level signal will not initiate the reactor or turbine trip. The DAS high hot leg temperature signal is needed to perform this function.

The staff evaluated ATWS for the AP1000 as documented in NUREG-1793. For that evaluation, the applicant analyzed a number of cases that included scenarios with and without normal feedwater operating. The most limiting case was confirmed to be the loss of normal feedwater event with turbine bypass operable, resulting in the highest RCS pressure. Addition of the hot leg temperature DAS trip would not alter that conclusion. The additional trip provides additional margin for the limiting case, and hence, is a conservative change that is acceptable for ATWS response. With respect to DAS reliability, quality, qualification, and independence from the primary protection system, the staff finds that the addition of a new function would not impact any of these design characteristics. Specifically, the function can be added into the current DAS architecture without changes to the DAS architecture described in the certified AP1000 design. Therefore, the staff finds the new DAS high hot leg temperature reactor and turbine trip meets the requirements of 10 CFR 50.62.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.8.3 Evaluation Findings and Conclusions

After reviewing WCAP-17184-P, Revision 2; TR-97, Revision 1; WCAP-15775, Revision 4; changes to the AP1000 DCD, Tier 1 and Tier 2; and responses to related RAIs, the staff concludes that the applicant provides sufficient information to support the changes made to the DAS in Revision 19 of the AP1000 DCD.

The staff concludes that the proposed changes to the AP1000 DCD meet the requirements of 10 CFR 50.62 and 10 CFR Part 50, Appendix A, GDC 22. The proposed changes to the AP1000 DCD did not affect the design commitments regarding an ATWS. The staff also found that the applicant provided sufficient information to support the removal of the two completed lifecycle phases and other changes for DAS. Therefore, the staff concludes the changes the applicant made to the DAS design meet regulatory requirements and criteria.

7.9 Data Communication Systems

7.9.1 System Description

The AP1000 I&C systems consist of the PMS safety system, which contains four independent divisions, four nonsafety systems, the PLS, data display and processing system (DDS), main turbine control and diagnostic system, and special monitoring system (SMS)), and two systems that perform both safety and nonsafety functions (IIS and operations and control centers system (OCS)). AP1000 DCD, Section 7.1.2.8, references WCAP-16675, Revision 5, for AP1000 data communications systems design. This TR provides an overview of the design of data communications within the PMS and communications between the PMS and nonsafety systems. Section 3 of WCAP-16675 references WCAP-16674, Revision 4, for a more detailed description of data communications in the AP1000 I&C systems design.

WCAP-16675, as supplemented by WCAP-16674, describes the use of the Common Q platform for data communications within the internal PMS safety functions (nuclear instrumentation system (NIS), QDPS, RTS, ESFAS, and the component logic system) and the safety portions of the IIS and OCS. This TR also describes the use of the Emerson Ovation® Network for data communications within the nonsafety systems, the nonsafety portions of IIS and OCS, and outputs from the safety system data (via the Advant-Ovation® Interface (AOI) gateways).

7.9.2 Communication within Safety Systems

7.9.2.1 Common Q Communications Subsystems

The NRC evaluated WCAP-16097, "Common Qualified Platform Topical Report," and issued safety evaluations approving the Common Q platform on August 11, 2000; June 22, 2001; and April 2, 2003. WCAP-16097 described the communications subsystems of the Common Q system. The SER for the Common Q platform provided an evaluation of the following three types of communications subsystems:

(1) AF100 bus communication for intrachannel communications and a separate AF100 bus for interchannel communications in the DDS

(2) HSL serial communications for interchannel communications

(3) external communications for communications between the Common Q platform and external computer systems

The Common Q platform topical report SER concluded that these three types of communications subsystems met the requirements of IEEE Std. 603-1991, as supplemented by IEEE Std. 7-4.3.2-1993. The staff's review of these communications subsystems supplements the conclusions made in the SER of the Common Q platform topical report. Specifically, the staff evaluated the application of these communications subsystems for data communications within the AP1000 I&C systems. Sections 7.9.2.2, 7.9.2.3, and 7.9.3 document the evaluation of the application of these communications subsystems to the PMS design.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC and satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.9.2.2 Intradivisional Communication via the AF100 Bus

7.9.2.2.1 Functional Description of the AF100 Bus

Section 3 of WCAP-16775 describes the use of the AF100 bus for intradivisional communications between the AC160 controllers and the safety and QDPS display systems within the same division. This section states that, within each PMS division, the internal functions and the safety portions of both IIS and the OCS are integrated using an intradivisional AF100 bus.

Specifically, the AF100 bus is used to allow the various AC160 controllers and FPDS within a division to exchange information for maintenance, test, diagnostic, communication (to the nonsafety system), display, and manual control. The majority of the dataflow is from the AC160 controllers to the FPDS (for display and for communication to the nonsafety system). Therefore, the AF100 bus is used to integrate information exchange among the AC160 controllers performing the ESFAS and reactor trip functions and the FPDS. The AF100 is not in the sensor-to-reactor trip path or sensor-to-ESFAS-actuation path. The ESFAS and reactor trip functions do not require information from each other to perform their safety functions.

The AF100 bus is a deterministic communication bus with a transmission rate of 1.5 megabit (Mbit)/second or faster. Section 4.1.2 of WCAP-16674 describes the two types of data communication that occur within the AF100 bus. The real-time data distribution communication provides a scheduled periodic broadcast of real-time data (process data transfer) pertaining to the plant processes. The general purpose communication provides a periodic exchange of data (message transfer) for other purposes, such as system operation, diagnostics, and maintenance. As described in Section 3.1 of WCAP-16675, message transfer does not influence process data transfer in any way. This is accomplished by reserving bandwidth for process data transfer and using the remaining bandwidth for message transfers. In addition, Section 3.1.1 of WCAP-16675 describes how the application program configuration tool is used to limit the maximum number of process data transfer packets that can be transferred over the AF100 bus to prevent overloading it. Process data packets are transferred with fixed packet size and cycle time to ensure deterministic communications.

Section 2.2.6.2 of WCAP-16675 states that software changes can be accomplished in the AC160 in two ways. One way is to program the AC160 over the AF100 bus. Even though this network and the only programming source (the MTP) are totally contained within a division of the PMS, this mode of programming is prevented. This is accomplished by using the AC160 Function Chart Builder tool to configure the equipment to not accept AF100 bus programming. The other way to load software into the AC160 is by a serial connection between the division's MTP and the AC160. Within a division, a separate cable is permanently routed from the maintenance and test cabinet (MTC) to each cabinet containing an AC160 processor module. This configuration allows for software loading to any processor module within a division from the MTP. The software loading cable is normally disconnected on each end. To perform a software update, the cable (coming from the cabinet containing the target processor module) in the MTC is connected to the MTP. The opposite end of the software loading cable is connected to the target AC160 processor module and the software update is performed from the MTP.

Section 3.1 of WCAP-16675 states that an AF100 bus is totally contained within each division of the safety system. The physical extent of each AF100 bus is limited to its corresponding I&C equipment room, the MCR, and the raceways between the two. Onsite access is not provided in any other location. Offsite access to the four PMS intradivisional Common Q networks is not available. This section also states that, within the PMS, security is maintained, since the ability to remotely program the AC160 controllers and safety and QDPS display systems over the AF100 bus has been disabled in the PMS. Access to the PMS intradivisional Common Q network is only available from the MCR. Access is not available in any of the other operation and control centers.

7.9.2.2.2 Evaluation of the AF100 Bus

The staff finds the use of the Common Q AF100 bus acceptable for intradivisional communication within each PMS division. The AF100 bus only serves one division in each PMS division, with no direct connections to other divisions or nonsafety systems. Data from other divisions and nonsafety systems can only reside on the AF100 bus via the other components within the same division (e.g., integrated communications processor (ICP) through HSLs). In such cases, electrical and communications isolation is provided by the fiber-optic connection between the given component and other divisions, and the communications processor of that particular component, respectively. IEEE Std. 603-1991, Clause 5.6.1, requires independence between redundant portions of safety systems to the degree necessary to retain the capability to accomplish the safety function during and following any design-basis event requiring that safety function. In addition, IEEE Std. 603-1991, Clause 5.6.3, “Independence Between Safety Systems and Other Systems,” requires independence between safety and nonsafety systems, such that credible failures in, and consequential actions by, nonsafety systems shall not prevent the safety system from accomplishing its intended safety function. Based on the communications and electrical isolation present in the Common Q AF100 bus, the staff finds the independence requirements of IEEE Std. 603-1991, Clauses 5.6.1 and 5.6.3, are met.

The staff evaluated the access controls to the AF100 bus against the requirements of IEEE Std. 603-1991, Clause 5.9, as clarified by the guidance provided in DI&C ISG #4-HICRc. IEEE Std. 603-1991, Clause 5.9, requires the design to permit the administrative control of access to safety-system equipment. DI&C ISG #4-HICRc, Section 1, Point 10, states that safety division software should be protected from alteration while the safety division is in operation. Hardwired interlocks or physical disconnection of maintenance and monitoring equipment should prevent online changes to safety-system software. The staff has evaluated the access controls described in Section 3.1 of WCAP-16675, as discussed in the above section.

The staff finds the use of the AC160 Function Chart Builder tool to configure the equipment so that it does not accept programming over the AF100 bus to be acceptable in meeting the requirements of IEEE Std. 603-1991, Clause 5.9, by addressing Section 1, Point 10, of DI&C ISG #4-HICRc regarding restrictions to the online programmability of safety equipment. In addition, the staff finds that the access control provided for programming the AC160 controller over the serial software loading cable to the equipment provides additional assurance that unauthorized software modifications to the AC160 controller and to the safety and QDPS displays are prevented. Specifically, the staff finds the physical disconnection of the software load cable between scheduled software updates meets Section 1, Point 10, of DI&C ISG #4-HICRc.

The staff evaluated how the design of the AF100 bus addressed the system integrity requirements of IEEE Std. 603-1991, Clause 5.5, which requires in part that safety systems be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. BTP 7-21, "Guidance on Digital Computer Real-Time Performance," states that risky design practices such as non-deterministic data communications, non-deterministic computation, use of interrupts, multitasking, dynamic scheduling, and event-driven design should be avoided. Based on the deterministic nature of the process data transfer on the AF100 bus, as described in Section 3.1 of WCAP-16675, and the limitations on maximum allowed process data transfer packets, the staff finds that the design of the AF100 bus adequately addresses the deterministic communications criteria provided in BTP 7-21 to meet IEEE Std. 603-1991, Clause 5.5.

7.9.2.3 Interdivisional and Intradivisional Communication via the High-Speed Link

7.9.2.3.1 Functional Description of the High-Speed Link

Section 3 of WCAP-16775 describes the use of the HSL for interdivisional and intradivisional communication within the PMS. This section states that the PMS uses HSLs, which are point-to-point, to communicate certain data within and across PMS divisions. The HSL is a serial RS 422 link using high-level datalink control protocol with a 3.1 Mbits/second transfer rate. The HSL is used for planned data exchanges of predefined data packets between two Common Q processor modules in the sensor-to-reactor trip path or sensor-to-ESFAS actuation path.

As stated in Section 6.2.1 of the SER to WCAP-16097, the PM646 function processor is divided into two sections, the process section and the communications section. The communications section in the PM646A processor module is used for HSL communications. Each processor module has one independent transmit link (output to two ports) and two independent receive links. The receivers of each HSL are independent and can receive different data independently, in accordance with the guidance of DI&C ISG #4-HICRc. As specified in Section 5.1 of the SER to WCAP-16097, these ports are used with fiber-optic cables for interchannel communications. Section 6.4 of this TR states that the integrity of data transmitted is monitored by using a CRC. The receiving processor module calculates the CRC of the received data and compares it with CRC bits received with the data. If the CRC comparison fails three consecutive times, the processor module declares the link has failed and reports the failure to the application software, which takes appropriate action. The processing section and the communication section of each PM646A processor module communicate with each other, in accordance with DI&C ISG #4-HICR, such that the communication and function processors operate asynchronously, sharing information only by means of dual-ported memory or some other shared memory resource that is dedicated exclusively to this exchange of information. This allows the two sections to share data between them while preventing either from affecting the operation of the other. The specific implementation, as described in Section 6.2.1 of the SER to WCAP-16097, is proprietary.

Based upon the specification listed in Section 4.1.1.4 of WCAP-16097, to ensure deterministic behavior of the Common Q platform, the measured load of the application programs on a single PM646 processor has to be less than 70 percent. To verify that safety systems meet the response time requirement, WCAP-16097 states that the applicant committed to perform a throughput analysis and a response time analysis. This topical report stated that, during the testing phase of the Common Q application, the applicant will perform response time tests to validate the design's compliance with both the system response and the display response requirements. In the SER for the WCAP-16097, the staff concluded that the design features, the operation of the AC160 PLC system, and CENP's commitments to perform timing analyses and tests provide sufficient confidence that the AC160 will operate deterministically to meet the recommendations in BTP HICB-21 and is, therefore, acceptable in that regard. However, the staff issued PSAI 6.6 to ensure that timing analysis and validation tests for applications of Common Qualified platform system verify that the design satisfies the plant-specific requirements for accuracy and response time presented in the accident analysis in Chapter 15 of the safety analysis report. The resolution of PSAI 6.6 is documented in NUREG-1793 for the certified AP1000 design, which states: "The accuracy and response time of the AP1000 safety systems will be commensurate with the Chapter 15 safety analysis. The COL applicant is responsible for the setpoint analysis. The setpoint analysis shall be performed by the COL applicant, as defined in DCD Tier 1, Section 2.5.2, Item 10, and DCD Tier 2, Section 7.1.6. This is COL Action Item 7.2.7-1."

Section 3.2 of WCAP-16675 describes the functional use of the HSL for interdivisional and intradivisional communication within the PMS. Below is a summary of how the HSL is used for data communications between equipment within the PMS.

Bistable Processor Logic to Local Coincidence Logic Communication

The PMS uses the Common Q HSLs to transmit certain data for partial trip, partial actuation, and related status information calculated in the BPL controllers to the LCL controllers. In addition, these serial links are used to transmit voting information between divisions. Fiber-optic cables provide electrical isolation and the communications processor in the PM646 module provides communications isolation.

Local Coincidence Logic to Integrated Logic Processor Communication

The PMS uses Common Q HSLs to transfer ESF system-level actuation and related status information calculated in the LCL controllers to ILPs that distribute the signals to the safety components via the CIM sub-system of the PMS. These links are only used locally within a division.

Integrated Logic Processor Communication to Integrated Communication Processor Communication

The PMS uses Common Q HSLs to transfer data to support the QDPS function and data to support cross-division diagnostics between divisions. Cross-division diagnostics are completed outside the PMS, using outputs from the ICP to the PLS. For communications across divisions of the PMS, fiber-optic media converters and fiber-optic cables provide electrical isolation and the communications processor within each PM646 module provides communications isolation. Section 3.3 of WCAP-16775 states that qualified isolation devices maintain electrical isolation and communications independence between the ICP and the PLS.

Integrated Test Processor

The PMS uses Common Q HSLs to transfer data to the ITP to support testing and monitoring the PMS system. The ITP compares information from within the division via the AF100 bus to information received via HSLs from the other divisions for fault detection. Fiber-optic media converters and fiber-optic cables provide electrical isolation and the communications processor within each PM646 module provides communications isolation.

7.9.2.3.2 Evaluation of the High-Speed Link and PM646 Deterministic Performance

Clause 5.5 of IEEE Std. 603-1998 requires safety systems be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. In addition, Clause 4.10 of IEEE Std. 603-1998 requires, as a part of the design basis, identification of the critical points in time or the plant conditions, after the onset of a design basis event.

To meet IEEE Std. 603-1991, Clause 5.5 and Clause 4.10, data communications systems in support of the protection system should demonstrate real-time performance in accordance with NUREG-0800 BTP 7-21. NUREG-0800 BTP 7-21 stipulates that:

1. Time delays within the data communications systems and measurement inaccuracies introduced by the data communications systems should be considered when reviewing setpoints.

2. Data rates and data bandwidths should be reviewed including impact by environmental extremes.

3. Sufficient excess capacity margins should be available to accommodate future increases.

In addition, limiting response times should be consistent with safety requirements. Digital computer timing should be consistent with the limiting response times and characteristics of the computer hardware, software, and data communications systems.

As stated above, in the SER for the Common Q topical report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q topical report, and the resolution of PSAI 6.6 in NUREG-1793. The applicant included ITAAC acceptance criteria to verify the PMS response time under maximum central processing unit (CPU) loading meets Chapter 15 response time limits. Specifically, the applicant committed to modify the acceptance criteria in Item 11d) the system integration and test phase of ITAAC Table 2.5.2-8 in Tier 1, Chapter 2, Section 2.5.2 of the AP1000 DCD to state "Performance of system tests and the documentation of system test results, including a response time test will be performed under maximum CPU loading to demonstrate the PMS can fulfill its response time criteria." The staff finds this proposed update to Item 11d) of ITAAC Table 2.5.2-8 acceptable. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

IEEE Std. 603-1991, Clause 5.6.1, requires independence between redundant portions of safety systems to the degree necessary to retain the capability to accomplish the safety function during and following any design-basis event requiring that safety function. NUREG-0800 BTP 7-11, "Guidance on Application and Qualification of Isolation Devices," states that fiber-optic cables are acceptable isolation devices. Based on the guidelines in BTP 7-11, the staff finds that optical media converters with optical fiber cabling in the HSL provide adequate electrical isolation. In addition, DI&C ISG #4-HICRc clarifies existing guidance on acceptable methods to meet the communications independence requirements of IEEE Std. 603-1991, Clause 5.6. Section 1 of this ISG specifies that communications between redundant divisions of safety systems should adhere to the points presented in that section. Table 7.9-1 documents the staff's evaluation of interdivisional communications using the HSLs against each criterion in DI&C ISG #4-HICRc.

Based on the staff's evaluation of the HSL design against the 20 criteria presented in Section 1 of DI&C ISG #4-HICRc, the staff finds that this design has addressed all criteria in Section 1 of DI&C ISG #4-HICRc, as shown in Table 7.9-1. Thus, the staff finds that the PMS design meets the independence requirements of IEEE Std. 603-1991, Clause 5.6.1.

IEEE Std. 603-1991, Clause 5.6.3, requires independence between safety and nonsafety systems, such that credible failures in, and consequential actions by, nonsafety systems shall not prevent the safety systems from accomplishing their intended safety function. Section 7.9.3 discusses communications independence between the ICP and the PLS to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3.

The staff has evaluated the access controls to HSLs against requirements of IEEE Std. 603-1991, Clause 5.9. Clause 5.9 required the design to provide administrative control of access to safety-system equipment. Since the HSLs are point-to-point dedicated serial links that are only used within safety systems of the PMS, access control is maintained by the physical security controls in the MCR. The staff finds that access controls to the HSLs meet the requirements of IEEE Std. 603-1991, Clause 5.9.

7.9.2.4 CIM Communication

WCAP-17179, Revision 2, provides a description of the data communications for the CIM. Section 2.1 of this technical report states that the CIM is designed to interface a field component to the PMS and the PLS. Communication with the PMS is accomplished with the SRNC assembly. The SRNC module accepts a HSL connection. The SRNC communicates with each CIM through a safety bus known as the X bus. The X-bus is an independent, bidirectional link between the CIM and the SRNC. The PMS communication link is known as the X port. The CIMs communicate with the PLS through an Ovation® RNC. The Ovation® RNC bus is known as the Y bus.

Section 2.3.1.2.8 of WCAP-17179 states that the CIM has design features to provide for deterministic operation of the CIM. Communication between the PMS to the SRNC via the HSL is designed to be deterministic as described in Section 2.4.1.1 of this TR. Furthermore, the communication between the SRNC and the CIM via the X bus is designed for deterministic communications as described in Section 2.4.1.2 of this TR. Messages received from the PLS via the Y bus is translated to discrete digital signals prior to input into the CIM.

Evaluation of the CIM Communications

The staff evaluated how the CIM communications design addressed the system integrity requirements of IEEE Std. 603-1991, Clause 5.5, which requires in part that safety systems be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. BTP 7-21 states that risky design practices such as non-deterministic data communications, non-deterministic computation, use of interrupts, multitasking, dynamic scheduling, and event-driven design should be avoided. Based on the design commitments for deterministic operation and communications for the CIM, as stated in WCAP-17179 and summarized above, the staff finds that the applicant has adequately addressed the deterministic communications criteria provided in BTP 7-21 to meet IEEE Std. 603-1991, Clause 5.5.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
1	The staff finds Point 1 of DI&C ISG #4 has been satisfied.	Aside from voting purposes, the components within the PMS do not receive any information from outside their division that will be used for accomplishing any safety functions.
2	The staff finds Point 2 of DI&C ISG #4-HICRc has been satisfied.	Except for voting purposes, the information received from other divisions is not used for initiating a protective function. Information received from other divisions for display and signal comparison purposes is validated through CRC checks. A communications fault in which the information is delayed, incorrect, or missing will be handled by the communications processor of each PM646 module.
3	For Point 3 of DI&C ISG #4, the applicant provided an alternative method to the guidance. The staff finds the alternative method is acceptable.	Aside from voting purposes, the information shared across divisions of the PMS does not meet DI&C ISG #4-HICRc interdivisional communication Point 3. In the PMS design, information received from other divisions for display and cross-divisional diagnostic purposes does not enhance the safety function. However, since: (1) the received information is not used for reactor trip and ESF actuation function, (2) it is communicated on a separate communication medium (AF100 bus), and (3) failure of such communication would not affect the safety functions, the staff finds the proposed alternative method acceptable.
4	The staff finds Point 4 of DI&C ISG #4 has been satisfied.	Within the PM646 module, the communications processor is separate from the function processor, and data transmission between the two processors can only be accomplished using dual port-RAM. As such, any communications errors will not propagate from the communications processor to the function processor within each PM646 module.
5	The staff finds Point 5 of DI&C ISG #4 has been satisfied.	As stated above, in the SER for the Common Q Topical Report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q Topical Report, and the resolution of PSAI 6.6 in NUREG-1793. The PMS is designed to meet the Chapter 15 overall response requirements. The proposed response time testing in ITAAC Item 11d of Table 2.5.2-8 in the AP1000 DCD will verify that the as-built PMS fulfills the response time criteria under maximum CPU loading.
6	The staff finds Point 6 of DI&C ISG #4 has been satisfied.	The safety processor operates asynchronously to the communications processor within the PM646 module, and the operation does not perform any communication handshaking or accept interrupts from other divisions.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
7	The staff finds Point 7 of DI&C ISG #4 has been satisfied.	The communication section of PM646 module only accepts predefined data sets and uses a CRC check to ensure the integrity of the data.
8	The staff finds Point 8 of DI&C ISG #4 has been satisfied.	The received information from other divisions is only used for display, diagnostic, and voting purposes. The use of predefined data sets and CRC checks ensures that the integrity of the data prevents any communications errors from affecting the safety functions.
9	The staff finds Point 9 of DI&C ISG #4 has been satisfied.	Section 2.1 of WCAP-17201 describes how the incoming messages are pre-allocated memory in static locations in the communications portion of the AC160 controller. Additional features are implemented to ensure that any software faults in the application data will be detected. Based on the storage of incoming messages in pre-allocated memory space and the use of error detection features to identify any software faults, the staff finds that Point 9 has been satisfied.
10	The staff finds Point 10 of DI&C ISG #4 has been satisfied.	As stated in Section 7.9.2.2.2 of this report, there only two methods to load software to the AC160 controllers. One is via the AF100 bus, which has been disabled during the software programming. The other way to load software into the AC160 is by a serial connection between the division's MTP and the AC160. This loading cable is normally disconnected on each end to prevent inadvertent programming during operations. As such, the staff finds Point 10 has been satisfied.
11	The staff finds Point 11 of DI&C ISG #4 has been satisfied.	The information received from other divisions is only used for display, diagnostic, and voting purposes. The design does include using the information for any other functions, including functions that allow the safety function to receive software instructions. The implementation of the design will be evaluated in the review of the specific design specifications.
12	The staff finds Point 12 of DI&C ISG #4 has been satisfied.	WCAP-17201, Section 2.2, discusses how messages will be checked to ensure validity of the message (e.g., repeated messages or messages out of sequence.) The specific features implemented to ensure the validity of the messages are proprietary. However, based on the information presented in technical report WCAP-17201, Section 2.2, the staff finds that Point 12 of ISG-04 has been satisfied.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
13	The staff finds Point 13 of DI&C ISG #4 has been satisfied.	As described in Section 2.3 of WCAP-17201, the HSL protocol does not support error detection; only error detection is implemented to detect data communication failures. Once a message is detected as bad, it is flagged, and the application software will respond accordingly. The staff finds the use of error detection for HSL data communications acceptable to ensure received messages are correct and correctly understood. Therefore, the staff finds that point 13 has been satisfied.
14	The staff finds Point 14 of DI&C ISG #4 has been satisfied.	The HSL is a point-to-point serial link; thus, Point 14 of DI&C ISG #4 is satisfied.
15	The staff finds Point 15 of DI&C ISG #4 has been satisfied.	Although the Common Q topical report states that the communication section of the PM646 is event driven, and does not communicate a fixed set of data at regular intervals, the processing section of PM646 is cyclic. As such, as discussed in Section 2.4 of WCAP-17201, at the end of every execution cycle, the application program store data from the processing section into the dual-ported memory for the communication section to transmit. Since the communication section transmits the data whenever there is a message in dual-ported memory, this communication is thus linked to the cyclic operation of the process section. Therefore, although the communication section is event driven, the communications are really cyclic and deterministic, and thus the staff finds that Point 15 has been satisfied.
16	The staff finds Point 16 of DI&C ISG #4 has been satisfied.	The watchdog timer feature of the PM646 module ensures message and link liveliness.
17	The staff finds Point 17 of DI&C ISG #4 has been satisfied.	The AP1000 DCD and the supporting TRs, along with the Common Q topical report, and Tier 1 ITAAC have committed to completing equipment qualification.
18	The staff finds Point 18 of DI&C ISG #4 has been satisfied.	The AP1000 DCD and the supporting TRs, along with the Common Q topical report, have committed to providing the FMEA and the SHA.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
19	The staff finds Point 19 of DI&C ISG #4 has been satisfied.	As stated above, in the SER for the Common Q Topical Report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q Topical Report, and the resolution of PSAI 6.6 in NUREG-1793. The PMS is designed to meet the Chapter 15 overall response requirements. The proposed response time testing in ITAAC Item 11d of Table 2.5.2-8 in the AP1000 DCD will verify that the as-built PMS fulfills the response time criteria under maximum CPU loading.
20	The staff finds Point 20 of DI&C ISG #4 has been satisfied.	As stated above, in the SER for the Common Q Topical Report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q Topical Report, and the resolution of PSAI 6.6 in NUREG-1793. The PMS is designed to meet the Chapter 15 overall response requirements. The proposed response time testing in ITAAC Item 11d of Table 2.5.2-8 in the AP1000 DCD will verify that the as-built PMS fulfills the response time criteria under maximum CPU loading.

The evaluation of the interface between the CIM and the PLS with respect to communications independence requirements, as stipulated in IEEE Std. 603-1991, Clause 5.6.3, is provided in Section 7.9.3 of this report.

7.9.2.5 Main Control Room Multiplexers

Section 7.1.2.6 of the AP1000 DCD removed the use of multiplexers in the protection and safety monitoring system to provide a signal path between the protection system equipment and the MCR. This section states that each division's safety and QDPS display will communicate with the protection system equipment via the dedicated AF100 communications network within each division. In addition, Section 3.4.1 of WCAP-16675 states that the MCR system-level actuation switches are cabled directly from the switches in the MCR to the LCL located in the bistable coincidence cabinets in each instrument room.

Since the switches in the MCR are directly connected to the LCL, the staff finds the justification for removal of the multiplexers in the MCR acceptable.

7.9.2.6 Testing of Communications Modules

The applicant removed the description of the fault tolerances, maintenance, test, and bypass from the DCD, and replaced it with references to WCAP-16675. Section 6.1 of WCAP-16675 describes the test features of communications modules within the PMS.

Sections 6.1.2 of WCAP-16675 states the AF100 bus communication modules provide communications between subsystems (e.g., BPL, LCL, ILP, MTP, and ITP). These communications include transferring data in support of system diagnostics. The AF100 bus supports two types of communications: process data and message transfer. Process data are dynamic data used to monitor and control the process, while message transfer is used for program loading and system diagnostics.

The AF100 bus communications modules are individually supervised by their own internal diagnostics and additional run-time diagnostic. In addition, the processor module performs continuous background diagnostics of the communications modules and automatically detects errors during operation. The process module contains the error messages in the error buffer for system troubleshooting.

Evaluation for Testing of Communications Modules

IEEE Std. 603-1991, Clause 5.7, requires the design to provide the capability to test and calibrate safety-system equipment, while retaining the capability of the safety systems to accomplish their safety functions. As applied to data communications systems, NUREG-0800 Section 7.9 states that data communications systems should be designed to support self-testing and surveillance testing. The design of automatic self-test features should maintain channel independence. The staff finds that the self-testing of the AF100 bus communications modules, including the internal diagnostics and additional runtime diagnostics, demonstrate conformance with the self-testing criteria in Section 7.9 of NUREG-0800 and, thus, meets the requirements of IEEE Std. 603-1991, Clause 5.7. However, the staff notes that these self-test features are not to replace the requirements for surveillance testing. In addition, since self-testing of the AF100 bus does not traverse multiple divisions or go outside the safety system, these testing features meet the independence requirements of IEEE Std. 603-1991, Clause 5.6.

7.9.3 Communication between Safety and Non-safety Systems

Section 5 of WCAP-16674 describes the data communications between the safety system and nonsafety system within the AP1000 design. This TR also describes the changes made to the certified AP1000 design regarding data communications between the safety and nonsafety systems. This TR states that the certified AP1000 design had the following data flow between the safety and nonsafety systems:

(1) data flow from PMS to PLS for control purposes
(2) data flow from PMS to DDS for information system purposes
(3) data flow from DDS to PMS for safety-system actuation purposes (via Remote Shutdown Panel when activated)
(4) data flow from PLS to PMS for component control purposes

In addition, the certified design establishes the following ITAAC in AP1000 DCD Tier 1, Section 2.5.2, regarding the implementation of these data flows:

7.a The PMS provides process signals to the PLS through isolation devices.

7.b The PMS provides process signals to the DDS through isolation devices.

7.c Data communications between safety and nonsafety systems does not inhibit the performance of the safety function.

7.d The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

In the certified design, the data flow between safety and nonsafety systems is primarily implemented using divisionalized bidirectional gateways. Section 5 of WCAP-16674 states that the data communications between the PMS and the nonsafety system have been modified in the PMS design. These modifications have the following effects:

- Reduce the dependence on the gateways.
- Make the gateways [ ].
- Create segmentation and network independence of the nuclear steam supply system (NSSS) control functions within the PLS.
- Make a clear delineation of the points of electrical, communication, and functional isolation.

In the modified design, the PMS implements data flows between safety and nonsafety equipment using divisionalized, [ ] gateways and individual analog and digital signals. Five cases of safety-system-to-nonsafety-system communication are identified within the AP1000 design. WCAP-16674 provides an analysis of the ITAAC in AP1000 DCD Tier 1, Section 2.5.2 for compliance for each of the cases, including the following requirements:

- There are isolation devices between the PMS and PLS, and between the PMS and the DDS.
- Data communications between safety and nonsafety systems do not inhibit the performance of the safety function.
- PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC to satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.9.3.1 Description of the Five Cases of Communication between Safety and Nonsafety Systems

Below is a summary of the five cases of safety-system-to-nonsafety-system communication.

Case A and Case B

Section 5 of WCAP-16674 states that Case A and Case B communications allow the PMS to communicate with the nonsafety control system (PLS) via qualified isolation devices. Case A involves transferring safety-related input signals that are isolated in the PMS cabinets and sent to the PLS as individual hardwired analog signals. This is identical to the type of interface in existing plants. Case B allows the PMS to transfer analog and discrete digital signals calculated within the PMS to the PLS using qualified isolation devices. Section 5 of WCAP-16674 states that the qualified isolation devices used in both Case A and Case B communications provide electrical isolation between the systems as required by IEEE Standard 603-1991. They also provide functional isolation by preventing the nonsafety system from adversely affecting the safety function.

Case C

Case C communications allow various process-related signals (analog input signals, analog signals calculated within the PMS, and digital signals calculated within the PMS) to be sent to the DDS for information system (plant computer) purposes. Non-process signals, such as cabinet entry status, cabinet temperature, and direct current power supply, are also provided to the DDS for information system purposes. As described in Section 5.1.2 of WCAP-16674, the AOI Gateway in each PMS division connects that division's internal network to the nonsafety real-time data network. The sole purpose of the AOI Gateway is to provide data from the safety system to the nonsafety system for nonsafety applications. The AOI Gateway has no protection function in the PMS. The reliability of the PMS to perform its safety function is not dependent on the AOI Gateway's being functional.

The gateway has two subsystems: one is the safety subsystem that interfaces with the AF100 bus, and the other is the nonsafety subsystem that interfaces with the nonsafety Emerson Ovation® network. The AOI safety subsystem is implemented within the PMS to gain access to the desired data. This functionality is included in the PMS MTP, a Common Q FPDS. The PMS portion of the AOI function is implemented using the hardware and software dedication and qualification methodologies accepted by the NRC as part of the Common Q topical report SER process. A fiber-optic link provides communication between the safety subsystem and the nonsafety subsystem. Communication isolation is achieved through the use of [ ] transmission of data from the optical transmitter on the safety subsystem to the optical receiver on the nonsafety subsystem.

For sequence of events (SOE) signals, such as partial trip signals, reactor trip signals, and ESFAS, each division provides the signals to the SOE system or interface via a [ ] fiber-optic link. The flow of information is strictly from the safety subsystem to the nonsafety SOE system or interface. The [ ] nature of the link is assured by the use of a single [ ] fiber. The safety end of the fiber is connected to an optical transmitter. The nonsafety end of the fiber is connected to a fiber-optic receiver. This arrangement also provides electrical isolation between the safety and nonsafety portions of the system.

<u>Case D</u>

Case D communications allows the nonsafety system to communicate with the safety system using discrete digital signals. These signals are used to implement nonsafety manual control of system-level safety functions (actuations, manual blocks, and resets, manual reactor trip) and a nonsafety interlock of certain PMS test functions.

Case D communications allows nonsafety manual controls of system-level safety functions that originate from dedicated switches in the RSR. Section 5.2.1 of WCAP-16674 states that in the RSR, the nonsafety manual controls of system-level safety functions (actuations, manual blocks and resets, manual reactor trip) originate from dedicated switches. The individual discrete digital signals are classified as nonsafety-related and are, therefore, isolated in the PMS cabinets before being used. At the RSR, a fiber-optic transmitter encodes the switch contact state to send over the fiber-optic cable. In the PMS, the fiber-optic receiver decodes the data and recreates the switch contact state on its discrete output signal to the AC160 rack in the safety system. Electrical isolation is provided via the fiber-optic connection. There is no metallic path to conduct an electrical fault into the PMS. Functional isolation is provided by logic within the PMS to prevent the nonsafety data flow from inhibiting the safety function. The functionality associated with these controls is disabled until operation is transferred from the MCR to the RSR. This transfer is accomplished by the divisionalized Class 1E transfer switches, which are connected directly to the LCL controllers in each division. Additionally, when the controls are enabled, their functionality is limited to that defined in the PMS functional design, because the information transferred is only in the form of discrete digital signals (i.e., there is no computer software-based communication). Specifically, the PMS design only permits the RSR manual system-level ESF actuations and the manual reactor trip inputs to initiate safety functions, not inhibit them. The manual system-level resets only remove the system-level actuation signals; they do not cause any components to change state. An additional signal is required to cause a component to change state. To reduce the chance of the spurious actuation of a function, switch contacts and communication paths are arranged in complementary pairs. Two simultaneous failures in opposite directions would be required to cause a spurious actuation.

In addition for some PMS test functions that are subject to interlocks, Case D communications also allows for transfer of discrete individual hardwired digital signals for interlocks from nonsafety equipment to the PMS. Section 5.1.2 of WCAP-16674 states that, for certain PMS test functions that are subject to interlocks from nonsafety equipment, individual hardwired digital signals from nonsafety systems are isolated in the PMS cabinets before being used. Qualified isolation devices are used. These devices provide electrical isolation between the systems, as required by IEEE Std. 603-1991.

Case E

Case E communications allows the nonsafety system to communicate with the safety system using discrete digital signals. These signals are used to implement nonsafety manual component-level controls of safety components.

As described in Section 5.2.2 of WCAP-16674, Case E communications allows manual component soft controls originating in the PLS to actuate safety components. The use of a remote I/O node, consisting of one or more Class 1E CIMs, will congregate the signals from the nonsafety manual soft controls to provide one signal digital output to a non-processor-based priority logic also contained in the CIM. The remote I/O node from the nonsafety system is physically located within each division of the safety system. The remote I/O node is electrically isolated from the nonsafety system by the fiber-optic remote I/O bus. The node is powered by the safety system, and the portions of the node not performing a safety function are qualified as an associated circuit, in accordance with IEEE Std. 384-1981. Specifically, the safety-system qualification program will demonstrate that, when it is subject to environmental, electromagnetic, and seismic stressors, it does not degrade the Class 1E circuits below an acceptable level. The environmental, electromagnetic, and seismic stressors used for these tests are the same as those used to qualify the Class 1E equipment in the same cabinet.

Within the CIM, demands from the nonsafety system are evaluated against Class 1E automatic actuation signals and Class 1E manual actuation signals from the PMS subsystem. If conflicting demands are present, the safe state of the component takes priority. The CIM uses non-processor-based priority logic hardware to implement this priority function. The CIM module also provides status updates of the safety component to the PLS. The remote I/O bus that connects the remote I/O node to the PLS uses fiber-optic cables to provide electrical isolation. As depicted in Figure 6-3 of WCAP-16674, the remote I/O bus uses bidirectional communications between the PLS and the remote I/O node. However, the communications interface of the CIM translates this data into simple discrete signals for input into the Class 1E priority logic to ensure communications independence.

PMS Interfaces to Standalone Systems

In addition to the five cases of communications between the PMS and nonsafety systems, Section 4.2.1 of WCAP-16674 states that the PMS interfaces to the standalone radiation monitoring system (RMS). RMS has two parts, one for safety functions and the other for nonsafety functions. There is no interface between the two parts. The safety portion of the RMS interfaces with the PMS using simple analog and/or discrete digital signals; this interface does not use network or datalink connections. Communication isolation does not apply to discrete hardwired signals. Electrical isolation between the RMS and the PMS is not required since the safety portion of the RMS is Class 1E. There is no interface between the nonsafety portion of the RMS and the PMS.

The CETs used by the QDPS function of the PMS are physically housed within IIS. There is no electrical interface between the CETs and the incore instrumentation electronics of the IIS. The CETs interface to the PMS using simple analog signals; these interfaces do not use network or datalink connections. Communication isolation does not apply to discrete hardwired signals. Electrical isolation between the CETs and the PMS is not required since the CETs are Class 1E.

7.9.3.2 Evaluation of Safety to Nonsafety Data Communication

IEEE Std. 603-1991, Clause 5.6.3, requires independence between safety and nonsafety systems, such that credible failures in and consequential actions by nonsafety systems shall not prevent the safety system from accomplishing its intended safety function. In addition, GDC 24 requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel, which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements for the protection system. NUREG-0800 Section 7.9 provides acceptance criteria for independence between safety and nonsafety systems to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. NUREG-0800 Section 7.9 states that physical, electrical, logical, or software malfunction in one portion cannot adversely affect the safety functions of the connected system. In addition, NUREG-0800 BTP 7-11 provides acceptable methods for ensuring electrical isolation between safety and nonsafety systems. The staff evaluated each of the five cases of communication between the components within the PMS and nonsafety systems, and the PMS interface to nonsafety standalone systems, against the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. DI&C ISG #4-HICRc clarifies existing guidance on acceptable methods to meet the communications independence requirements of IEEE Std. 603-1991, Clause 5.6. Section 1 of this ISG specifies that communications between safety and nonsafety systems should adhere to the points presented in that section. The staff evaluated the safety-to-nonsafety communications scheme for each of the five cases against the criteria presented in DI&C ISG #4-HICRc. The staff's evaluation is documented below.

<u>Case A and Case B</u>

Based on the staff's evaluation of WCAP-16675 and WCAP-16674, the staff finds that the design of hardwired interfaces used to send analog and digital signals from the PMS to the PLS meets the electrical isolation and communications independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff finds that since the signal transmission between the PMS and PLS is limited to analog and discrete digital signals, communications independence do not apply. Section 5 of WCAP-16674 qualified isolation devices are used in Case A and Case B communications to provide electrical isolation between the PMS and the PLS. As stated in Section 7.1.2.10 of the AP1000 DCD, isolation devices are used to maintain the electrical independence of divisions, and to prevent interaction between nonsafety-related systems and the safety-related system. Isolation devices are incorporated into selected interconnections to maintain division independence. Isolation devices serve to prevent credible faults (such as open circuits, short circuits, or applied credible voltages) in one circuit from propagating to another circuit. The staff finds these design criteria are consistent with the guidance of NUREG-0800 BTP 7-11. Since these design criteria have not changed from the certified Revision 15 of the AP1000 DCD to the current revision, the staff finds the use of qualified isolation devices acceptable to ensure adequate electrical isolation between the PMS and nonsafety systems. Therefore, the staff finds that the applicant has satisfied the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. In addition, in Tier 1, Chapter 2, Table 2.5.2-8 of the AP1000 DCD, the staff identified ITAAC for isolation devices from the PMS to the PLS and from the PMS to the DDS to ensure electrical isolation. Since these ITAAC were approved in Revision 15 of the AP1000 DCD, and no modifications were made in subsequent revisions, the staff finds that these ITAAC adequately verify electrical isolation between the PMS and the PLS and DDS to meet the electrical isolation requirements in IEEE Std. 603-1991, Clause 5.6.3, GDC 24.

<u>Case C</u>

The staff evaluated the description of the safety-to-nonsafety system communications in Case C against the electrical isolation requirements, and the communications and functional independence requirements, of IEEE Std. 603-1991 and GDC 24. Based on the information presented in Section 5.1.2 of WCAP-16674, the staff finds the use of one-way fiber-optical communication between the MTP and the AOI Gateway, and between the PMS and the SOE system and interface in Case C, provides adequate electrical isolation and communications independence between the PMS and the nonsafety, Ovation® network to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff finds that the use of fiber-optic cable for electrical isolation is in accordance with BTP 7-11. The staff finds that communications independence is achieved in the design, since the design does not include an optical receiver on the MTP for data to traverse from the nonsafety network to the PMS. Since the communication is physically [ ] from the safety system to the nonsafety system, a failure within the nonsafety system cannot propagate to the safety system. The staff finds that the safety-to-nonsafety system communications in Case C meet the communication and functional independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24.

<u>Case D</u>

The staff evaluated the description of interconnections between the RSR and the PMS in support of nonsafety manual controls of system-level safety functions in Case D communications based on the electrical isolation and functional independence requirements of IEEE Std. 603-1991, Clause 5.6, and GDC 24. Based on the information provided in Section 5.2.1 of WCAP-16674, the staff finds that the design provides adequate electrical isolation, and communications and functional independence for manual system-level ESF actuation and manual reactor trip inputs from the RSR to the PMS to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff makes the following findings:

- The use of fiber-optic cables provides adequate electrical isolation, as specified in NUREG-0800 BTP 7-11.

- Since the data flow from the RSR to the PMS is in the form of discrete digital signals (e.g., no communications protocol or handshaking), the guidance in DI&C ISG #4-HICR does not apply. The use of discrete digital signals for initiating system-level ESF actuation and reactor trip from the RSR using point-to-point fiber cabling provides adequate communications independence between the PMS and the RSR.

- The software within the PMS, which allows the discrete signals to only initiate safety functions, specifically, initiation of system-level ESF actuation and reactor trip, provides adequate functional isolation.

The staff evaluated the description of data communications between nonsafety equipment and the PMS in support of certain PMS test functions that require interlocks in Case D communications based on the electrical isolation and communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6, and GDC 24. The staff finds that the information presented in Section 5.2.1 of WCAP-16674 provides an adequate description of how electrical isolation, communications and functional independence are achieved for the inputs from nonsafety equipment to the PMS to meet the requirements in IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff makes the following findings:

- Electrical isolation is provided between the PMS and nonsafety equipment through the use of an isolation device. As stated above in the evaluation of Case A and Case B communications, the isolation device serve to prevent credible faults (such as open circuits, short circuits, or applied credible voltages) in one circuit from propagating to another circuit, which is consistent with the guidance of NUREG-0800 BTP 7-11.

- Since the data flow from the nonsafety equipment to the PMS is in the form of discrete digital signals (e.g., no communications protocol or handshaking), the guidance of DI&C ISG #4-HICR does not apply. The use of discrete digital signals for activating interlocks for certain PMS tests provides adequate communications independence between the PMS and the nonsafety equipment.

- The software in the PMS allows the discrete signals to only affect the ability to perform tests. The interlocks do not affect automatic or manual safety functions.

In Tier 1, Chapter 2, Section 2.5.2, Table 2.5.2-8 of the AP1000 DCD, the staff identified ITAAC for isolation devices between the PMS and the PLS and between the PMS and the DDS to ensure electrical isolation. However, the staff did not identify any ITAAC for verifying electrical isolation between the PMS and the nonsafety equipment that will be used to activate interlocks for these PMS tests. The staff requested the applicant to provide additional information to demonstrate how the qualified isolation devices provide electrical isolation between the nonsafety equipment and the PMS. Specifically, the staff requested the applicant to provide an additional ITAAC to verify electrical isolation between the PMS and the nonsafety equipment to activate these interlocks. In its response letter, dated February 8, 2010, the applicant proposed to include an additional ITAAC in Tier 1, Chapter 2, Section 2.5.2, Table 2.5.2-8 of the AP1000 DCD. The proposed ITAAC states:

> The PMS receives signals from non-safety equipment that provide interlocks for PMS test functions through isolation devices.

The proposed acceptance criterion state:

> A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.

The staff finds the proposed ITAAC acceptable in verifying that adequate electrical isolation exists to prevent credible faults from nonsafety equipment from impacting the PMS, and thus satisfies the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. In Revision 19 to the AP1000 DCD, the applicant added Design Commitment 7e to Table 2.5.2-8, which resolves this issue.

<u>Case E</u>

The staff evaluated the description of data communications between the CIM and the PLS in Case E, based on the electrical isolation and communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6, and GDC 24. The staff finds that the information presented in Section 5.2.2 of WCAP-16674 provides an adequate description of how the applicant achieves electrical isolation and communications and functional independence for nonsafety manual component-level control of safety components for Case E, to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff makes the following findings:

- The use of fiber-optic cables between the remote I/O bus that connects the remote I/O node to the PLS provides adequate electrical isolation, as specified in NUREG-0800 BTP 7-11.

- Although the remote I/O bus uses bidirectional communications between the PLS and the remote I/O node, the communications interface of the CIM translates these data into simple discrete signals for input into the Class 1E priority logic. Since the data flow into the Class 1E priority logic is in the form of discrete digital signals (e.g., no communications protocol or handshaking), the guidance in DI&C ISG #4-HICR does not apply. The use of discrete digital signals for initiating nonsafety manual component-level control of safety components provides adequate communications independence between the CIM and the PLS.

The priority logic within the CIM provides functional isolation by ensuring that the PMS has priority to actuate the safety component, such that the nonsafety signal cannot prevent the PMS from actuating the component. If the nonsafety system initiates an actuation command without the PMS initiating an actuation command, then the safe state of the component takes priority.

IEEE Std. 603-1991 defines associated circuits as non-Class 1E circuits that are not physically separated or are not electrically isolated from Class 1E circuits by acceptable separation distance, safety class structures, barriers, or isolation devices. IEEE Std. 384-1992, Clause 5.5, includes the classification and qualification of associated circuits. IEEE Std. 384-1992, Clause 5.5.3, states that associated circuits, including their isolation devices or the connected loads without the isolation devices, shall be subject to the qualification requirements placed on Class 1E circuits to ensure that the Class 1E circuits are not degraded below an acceptable level. Associated circuits need not be qualified for performance of function, since the function is non-Class 1E. The staff finds that Section 5.2.2 of WCAP-16674 adequately demonstrates how the RNC is qualified as an associated circuit to meet the requirements of IEEE Std. 603-1991, Clause 5.6, and IEEE Std. 384-1992. Specifically, the staff finds that the commitment, as part of the overall safety-system qualification program, to demonstrate that, when the RNC is subject to environmental, electromagnetic, and seismic stressors, it does not degrade the Class 1E circuits below an acceptable level, meets the associated circuit qualification requirements in IEEE Std. 384-1992, Clause 5.5.3.

PMS Interfaces to Standalone Systems

Section 4.2.1 of WCAP-16675 provides a description of the PMS interfaces to standalone safety systems. This section states that the PMS interfaces to the standalone RMS. However, there is no interface between the safety portion of the RMS and the nonsafety portion.

In addition, the CETs used by the QDPS function of the PMS are physically housed within the IIS. There is no electrical interface between the CETs and the incore instrumentation electronics of the IIS. The CETs interface to the PMS using simple analog signals; these interfaces do not use network or datalink connections.

Based on the information provided in Section 4.2.1 of WCAP-16675 regarding PMS interfaces with standalone systems, such as the RMS, the staff finds the design adequately demonstrates compliance with the communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, for the RMS, since there is no interface between the safety portion of the RMS and the nonsafety portion, communications independence does not apply. In addition, functional independence is achieved through the isolation of safety functions to only the safety portion of the RMS, and does not require interaction with the nonsafety portion to perform the intended safety functions. For the CETS used by the QDPS function of the PMS, the staff finds that since the CETS interface to the PMS using simple analog signals and these signals are Class 1E, communication independence does not apply. Since the CETS are Class 1E, the staff finds that electrical isolation between the CETS and the PMS is not required.

7.9.4 Nonsafety Communications

7.9.4.1 Description of the Nonsafety Communication Network

Nonsafety communications consist primarily of the nonsafety communication network and the nonsafety data link interface. The nonsafety communication network is implemented using the Ovation® network. This network uses unaltered Ethernet protocols, high-speed Ethernet switches, and full duplex cabling (fiber or copper shield twisted pair).

Section 3 of WCAP-16674 provides a detailed description of the AP1000 nonsafety communications system. The nonsafety communications network provides real-time data distribution and general purpose communications. Real-time data distribution is defined as the scheduled periodic broadcast of real-time data pertaining to the plant processes. The term "general purpose communications" is defined as the aperiodic exchange of data for other purposes, such as system operation, diagnostics, and maintenance.

The Ovation® network supports network standard communications protocols, such as Transmission Control Protocol/Internet Protocol and User Datagram Protocol/Internet Protocol for general purpose communications. Within the Ovation® system, general purpose communications based on standard protocols are used for aperiodic data, including file-type data transferred from the historian and plant databases to be presented at the HSI, plant informational data messages, alarm messages, and SOE messages to the plant historian for long-term historical storage. This communication occurs on the same physical media as the real-time periodic data, but it is implemented in such a way as to preserve the design philosophy of guaranteeing the real-time periodic data transmission without loss, degradation, or delay, even during plant upsets.

With respect to periodic data, the network is designed to support up to 200,000 point values per second, using a nominal percentage of the overall network bandwidth. The network load associated with periodic data origination is constant; it does not change during plant upset conditions. The Ovation® vendor has tested the network at the limit of 200,000 point values per second. A design goal is to limit the number of point values per second, to the extent possible. This will provide additional spare capacity and will result in a lower base load on the network. As the design is finalized, a firm number of point values per second will be determined. This will be used to calculate the base network load and, therefore, the network bandwidth available for aperiodic data communications. With respect to aperiodic data, the network load is variable but managed. The aperiodic data levels can be managed through careful system configuration. Alarm message data will be minimized, to the extent possible, by limiting the number of points subject to alarm checking and by carefully selecting alarm limits to minimize nuisance alarms. Network impacts associated with station staff in the main control area are somewhat limited by the number of operators and operator stations and by the number of engineers and engineering stations. In general, the network load from aperiodic data traffic is expected to be very small in relation to the overall bandwidth of the system. Analytical justification of network capacities will be reviewed for correctness. Based on the current evaluation of expected network traffic, the single network design will meet or exceed all system capacity and network loading requirements.

Storm control is configured on the Ovation® network to ensure that highway availability requirements are satisfied, given the possibility that a software or hardware malfunction, or a malicious network attack, would introduce a packet storm on the control system highway. Storm control is implemented with configuration settings provided by the switch operating system. In general, each port subject to storm control is configured with traffic ingress block and restoration settings. These values are typically a percentage of the total available bandwidth that can be used by the broadcast or multicast traffic. When traffic entering a port exceeds the predefined block value, packet forwarding on the port is blocked. Packet forwarding resumes when the traffic falls below the predefined "restore forwarding" setting. Storm control is put in place to protect the network from data storms produced as a result of atypical conditions, including hardware malfunctions, and errors introduced by humans. The thresholds are set on a per-port basis, so that native Ovation® traffic (e.g., periodic process point data, aperiodic alarm message traffic) will not activate the storm control function. In addition to the system storm control configuration installed on the network switches, the Ovation® controller has been hardened against excessive network traffic through the use of a software modification that prioritizes critical control functionality over network communications.

7.9.4.2 Evaluation of the Non-safety Communication Network

The staff evaluated the adequacy of the nonsafety Ovation® network to perform the required control functions specified in the AP1000 DCD and the supporting TRs, as well as WCAP-16675 and WCAP-16674, including how the applicant met the requirements of 10 CFR 52.47(a)(9). Regulations in 10 CFR 52.47(a)(9) require applications for light-water-cooled NPPs to evaluate the standard plant design against the NUREG-0800 revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the NUREG-0800 acceptance criteria. NUREG-0800 Section 7.9 provides performance criteria for data communication systems; specifically, for system capacity, data rates, and bandwidth requirements. The staff finds that the description provided for system capacity, data rates, and bandwidth requirements, and the analysis on expected network traffic, presented in Section 3 of WCAP-16674, adequately address the performance criteria for data communications systems specified in NUREG-0800 Section 7.9. Specifically, the staff finds that the evaluation of expected network traffic demonstrates that the network design is bounded by the Ovation® system capacity and network loading requirements.

Specifically, the staff evaluated the nonsafety Ovation® network design features to determine how the applicant has addressed operating experience with data storm, as described in NRC Information Notice 2007-15, "Effects of Ethernet-Based, Non-Safety Related Controls on the Safe and Continued Operation of Nuclear Power Stations," dated April 17, 2007. The staff finds that the storm control features within the Ovation® network design adequately demonstrate how data storms are precluded in the Ovation® network. Specifically, the staff finds that the use of storm control configuration settings provided by the switch operating system, and the software feature that prioritizes critical control functionality over network communications, ensure that the

Ovation® controller can continue to control critical plant operations during a network storm or a complete loss-of-network event.

7.9.4.3 Description of the Non-safety Data Link Interfaces

Section 3.2 of WCAP-16674 describes the nonsafety data link interfaces in the AP1000 I&C design. Each system is summarized below.

Standalone Systems

The Ovation® system supports standard and custom data links, both at the controller and workstation level. Controller-level interfaces include standard interfaces to Allen-Bradley programmable logic controllers and GE Mark V/VI, Toshiba, and MHI turbine control systems, as well as a standard MODBUS interface and OSI PI historian interface. At the controller level, the data link interface can be accomplished via a standard I/O module (the R-line Link Controller), or via fast Ethernet communications interfaces at the controller processor level.

Remote I/O

The Ovation® system supports the use of remote I/O, so that I/O modules can be clustered close to field devices, minimizing field cabling costs and also accommodating harsher environments. Remote I/O is in contrast to local I/O, which is housed in the same cabinet as the controller or next to it in an extended cabinet. For local I/O, all I/O modules reside in up to four cabinets, which are placed side by side. All field wiring leads to these cabinets.

Non-safety Smart I/O Field Buses

The Ovation® system supports HART I/O, FOUNDATION™ Fieldbus, Prefabs DP, and DeviceNet™ smart I/O interfaces. The Ovation® fieldbus solution is modular, and a single controller can simultaneously interface to fieldbus devices, HART I/O modules, conventional I/O modules, and third-party I/O.

HART I/O

The Ovation® controller supports native HART I/O modules. HART is technology that provides a digital information signal superimposed on a 4-20 milliampere traditional sensor loop. The digitized signal provides up to four HART multivariables, which provide additional information from HART-enabled devices, eliminating additional cabling required to provide the same information using traditional sensors and control output devices.

The Ovation® HART input module has eight inputs, with each input having an individual HART modem (supporting up to four HART multivariables), and individual channel-to-channel isolation. The Ovation® HART output module has four channels, also with individual HART modems per channel, and individual channel-to-channel isolation.

Foundation Fieldbus

FOUNDATION™ Fieldbus H1 is typically used for analog devices, such as sensors and modulating control valves. A large assortment of “smart” devices is available with the interface. The Ovation® FOUNDATION™ Fieldbus solution is modular and scalable. The interface between the FOUNDATION™ Fieldbus instrumentation and the Ovation® controller is via dedicated, redundant Fieldbus Ethernet switches, and Ovation® FOUNDATION™ Fieldbus Gateways. There are up to 16 FOUNDATION™ Fieldbus gateways per controller, with up to four H1 segments per gateway and up to 16 devices per segment.

Profibus DP

Profibus DP is typically used for digital on/off devices. In addition to being supported by the appropriate devices, it is suitable for long distances while remaining less sensitive to power, grounding, polarity, and resistance concerns.

The Ovation® Profibus Interface uses a standard Ethernet switch, attached to the Ovation® controller via the controller's standard MODBUS/TCP third-party I/O capability.

DeviceNet

DeviceNet™ is an interface for discrete actuators and sensors. The Ovation® DeviceNet™ interface has the same fundamental design as the Profibus DP interface, using a standard Ethernet switch, attached to the Ovation® controller via the controller standard MODBUS/TCP third-party I/O capability.

Asset Management

Another important component of the intelligent field interface solution is the Asset Management Solutions (AMS) suite of software. AMS software and the associated SNAP-ON applications is a suite of software solutions for streamlining all maintenance activities related to instrumentation and valves in a process plant. This package can be integrated into the Ovation® workstation and Ovation® controller to give the user direct access to all intelligent devices connected to the Ovation® I/O. With AMS integrated into Ovation®, digitized HART or FOUNDATION™, Fieldbus parameters such as valve position can be mapped to Ovation® process points that can be used anywhere they are required in the Ovation® distributed control system. AMS provides direct visibility from the Ovation® workstation to each “smart” device in the plant that is connected to Ovation®.

7.9.4.4 Evaluation of the Nonsafety Data Link Interfaces

The staff evaluated the nonsafety data link interfaces within the AP1000 I&C design. IEEE Std. 603 1991, Clause 5.6.3, requires independence between safety and nonsafety systems so that credible failures in, and consequential actions by, nonsafety systems shall not prevent the safety system from accomplishing its intended safety function. In addition, GDC 24 requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. These nonsafety data link interfaces do not communicate with any of the safety systems beyond the five cases of safety-to-nonsafety system data communications specified in WCAP-16674. As such, the staff finds that the electrical isolation, communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24 do not apply.

7.9.5 Secure Development and Operational Environment

10 CFR Part 50, Appendix A, GDC 21 requires, in part, that protection systems (or safety systems) must be designed for high functional reliability commensurate with the safety functions to be performed. Criterion III of Appendix B to 10 CFR Part 50 requires, in part, that quality standards must be specified and design control measures must be provided for verifying or checking the adequacy of design.

10 CFR 50.55a(h) requires that safety systems for NPPs must meet the requirements stated in IEEE Std. 603-1991. Clause 5.6.3 of IEEE Std. 603-1991 requires safety systems to be designed, such that credible failures in and consequential actions by other systems will not prevent safety systems from performing their intended safety functions. In addition, Clause 5.9 of IEEE Std. 603-1991 requires the design to permit the administrative control of access to safety system equipment. These administrative controls shall be supported by provisions within the safety systems, by provision in the generating station design, or by a combination thereof.

RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," Revision 2, provides a method that the NRC finds acceptable for complying with the Commission's regulations (i.e., 10 CFR Part 50, Appendix A, GDC 21, Criterion III of Appendix B to 10 CFR Part 50, and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9) for promoting high functional reliability, design quality, and security for use of digital computers in safety systems of NPPs. Security in this context refers to the establishment of an SDOE for digital safety systems by: (i) measures and controls taken to establish a secure environment for development of the digital safety system against undocumented, unneeded and unwanted modifications; and (ii) protective actions taken against a predictable set of undesirable acts (e.g., inadvertent operator actions or the undesirable behavior of connected systems) that could challenge the integrity, reliability, or functionality of a digital safety system during operations. RG 1.152, Revision 2, utilizes the waterfall life cycle phases to provide a framework for establishing digital safety system security guidance, as well as criteria for acceptability, in the development of high quality safety systems.

As proposed in the response to Open Item OI-SRP7.1-ICE-01, Section 7.1.2.14.1 of the AP1000 DCD will be revised to reference, APP-GW-J0R-012, hereinafter referred to as the "PMS Computer Security Plan," to demonstrate how computer security is incorporated into the design and development of AP1000 safety systems. This plan provides a description of the planning phase for the AP1000 PMS. This plan summarizes the quality standards and design control measures implemented to provide computer security and ensure that the PMS and CIM are designed for high functional reliability commensurate with the safety functions to be performed throughout the development phases of digital safety system lifecycle. These commitments include the design and development of security features and development controls for the PMS and CIM. Although the PMS Computer Security Plan does not officially commit to conformance to RG 1.152, Revision 2, this TR addresses the system security aspects of the Common Q platform, PMS application, and CIM from the concepts phase through the test phase to protect against non-malicious events that is consistent with the criteria provided by RG 1.152, Revision 2.

The safety evaluation for AP1000 I&C secure development and operational environment, provides a separate assessment of the PMS Computer Security Plan and includes official use only information. This separate safety evaluation forms the basis for the following conclusions.

- The identified vulnerabilities in the PMS design for the conceptual phase of the development life cycle, and the security capabilities that mitigate these vulnerabilities adequately address Regulatory Position C.2.1 in RG 1.152, Revision 2.

- The performance of additional V&V activities for the Common Q platform satisfies the criteria for identifying and mitigating vulnerabilities as specified in Regulatory Position C.2.1 of RG 1.152, Revision 2.

- Although the vulnerabilities of the PMS and CIM development process are general, these vulnerabilities were only based on the conceptual phase assessment, and these vulnerabilities encompass more detailed vulnerabilities that may be identified in later portions of the development process. Thus, the identified vulnerabilities are acceptable in specifying particular portions of the development process that are susceptible to unintended or inadvertent modification to the PMS or CIM while under development or to the development tools. As such, the applicant has adequately addressed the criteria in Regulatory Position C.2.1 of RG 1.152, Revision 2, for the PMS application and the CIM. In addition, the quality assurance program for both the development of the PMS and the CIM, and the V&V process are adequate to mitigate the identified vulnerabilities by identifying and preventing inadvertent changes to the PMS and CIM design during development.

- By precluding capabilities for remote access to the PMS during operations in the design and by ensuring one way data flow from the PMS to nonsafety systems (except for use of discrete digital or analog signal), the applicant has satisfied Regulatory Position C.2.1 in RG 1.152, Revision 2. In addition, the commitment to ensure that the isolated development infrastructure (IDI) is created to preclude remote access is sufficient to satisfy Regulatory Position C.2.1 in RG 1.152, Revision 2.

- Based on the PMS access control functional requirements, the safety to nonsafety interfaces requirements, and the commitment to ensure that proper human factors are considered during the development of the PMS design, these requirements adequately provide sufficient measures to protect the PMS from inadvertent operator actions or unpredictable behavior of connected systems during operations to address the criterion in Regulatory Position C.2.2.1 of RG 1.152, Revision 2, to define the security functional performance requirements.

- The Common Q Platform has adequately addressed the criteria in Regulatory Position C.2.2.1 of RG 1.152, Revision 2. Specifically, the Common Q platform software V&V, as described in Section 5.5.3 of the Common Q Software Program Manual, was reviewed and accepted by the NRC in the SER for the Common Q platform SPM. This includes the approval of the V&V process for use of pre-developed software within the Common Q platform. The V&V activities that were performed on the Common Q platform are adequate to ensure the integrity, reliability, and functionality of this platform for use in the AP1000 PMS application.

- By incorporating the security requirements into the overall system requirements, the PMS V&V process in accordance with the Common Q SPM is adequate to ensure the correctness, completeness, accuracy, testability, and consistency of the system security requirements. Thus, the applicant has met the criteria in Regulatory Position C.2.2.1 of RG 1.152, Revision 2.

- Based on the described security measures provided in the IDI, the commitment to assess and mitigate vulnerabilities in the IDI, and the quality assurance and V&V process described in the PMS Computer Security Plan, the applicant has adequately ensured that undocumented code or functions are precluded in the design to meet Regulatory Position C.2.2.2 of RG 1.152, Revision 2.

- By incorporating the security features as part of the overall system design, the design process described in Section 2.3 of the PMS Computer Security Plan is adequate to ensure that the system security requirements is accurately translated into specific design configuration items to meet Regulatory Position C.2.3.1 of RG 1.152, Revision 2. In addition, the additional security assessment completed during the design phase is adequate to ensure that any vulnerability that has not been identified during earlier phases of the development life cycle is captured and that the security features chosen in the conceptual phase are adequate.

- Based on the quality assurance and V&V process described in the PMS Computer Security Plan, the staff finds that the applicant has adequately ensured that undocumented code or functions are precluded in the PMS and CIM design to meet Regulatory Positions C.2.3.2 of RG 1.152, Revision 2. This is based on control of the design document revision process, storage of design process in an accessed controlled manner, and requirements traceability to ensure that all design features are traceable to requirements specifications.

- By incorporating the security features as part of the overall system implementation, the implementation process described in Section 2.4 of the PMS Computer Security Plan is adequate to ensure that the system design is accurately transformed into code, database structures, and related machine executable representations to meet Regulatory Position C.2.3.1 of RG 1.152, Revision 2. In addition, the additional security assessment completed during the implementation phase is adequate to ensure that the security controls chosen are adequate.

- Based on the commitment to secure the IDI, to perform testing and scanning to identify undocumented code and functions, and to follow the quality assurance and V&V process described in Section 2.1.2.2 of APP-GW-J0R-012, Revision 1, the applicant has adequately ensured that undocumented code or functions are precluded in the PMS and CIM implementation to meet Regulatory Positions C.2.4.2 of RG 1.152, Revision 2.

- Based on the commitment to perform integration, system, and acceptance tests where practical and necessary on the PMS security features, including testing of the system configuration, and the performance of additional vulnerability assessments to ensure that no new vulnerabilities are identified in the PMS, the applicant has adequately addressed Regulatory Position 2.5 of RG 1.152, Revision 2.

- Based on the commitment to secure the testing environment and to test the hardware architecture, external communication devices, and configurations for unauthorized pathways that affect system integrity, the applicant has adequately addressed Regulatory Position C.2.5.2 of RG 1.152, Revision 2.

Based on the staff's conclusions on the PMS Computer Security Plan, as discussed above, the staff finds that the applicant has sufficiently addressed the criteria in Regulatory Positions C.2.1 through C.2.5 of RG 1.152, Revision 2, to meet the requirements of 10 CFR Part 50, Appendix A, GDC 21; Criterion III of Appendix B to 10 CFR Part 50; and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9; as it relates to security and reliability of the PMS application and CIM. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

7.9.6 Evaluation, Findings, and Conclusions

The staff reviewed the revisions to Section 7.1 and the associated TRs of the AP1000 DCD against the regulatory requirements of a data communications system as stipulated in the guidance of NUREG-0800 Section 7.9. Below is a summary of the staff's findings.

Regulations in 10 CFR 50.55a(h) require compliance with IEEE Std. 603-1991 and the correction sheet, dated January 30, 1995. The minimum requirements that are applicable to all data communications systems are in IEEE Std. 603-1991, Clause 5.6.3. Other criteria include those in Clauses 5.4, 5.6.1, 5.7, and 5.9. The staff evaluated the data communications systems in the amendments to the AP1000 I&C systems design for conformance to the requirements of IEEE Std. 603-1991 and has the following findings:

- IEEE Std. 603-1991, Clause 5.4: This requirement has been fully satisfied, as documented in Section 7.9.3. In Table 2.5.2-8 of the AP1000 DCD, the staff identified ITAAC for the seismic, environmental, and Class 1E qualification of PMS equipment, including equipment used for data communications in the PMS.

- IEEE Std. 603-1991, Clause 5.6.1: This requirement has been fully satisfied, as documented in Section 7.9.2.3. The staff finds that the 20 criteria presented in Section 1 of DI&C ISG #4-HICRc have been satisfied in the design for interdivisional communication between divisions of the PMS.

- IEEE Std. 603-1991, Clause 5.6.3: This requirement has been fully satisfied, as documented in Section 7.9.3. The staff finds that the information presented in the AP1000 DCD and the supporting TRs have sufficiently demonstrated how independence is achieved for each of the five cases of safety and nonsafety communications.

- IEEE Std. 603-1991, Clause 5.7: This requirement has been satisfied, as documented in Section 7.9.2.5.

- IEEE Std. 603-1991, Clause 5.9: This requirement has been fully satisfied, as documented in Section 7.9.2. The AP1000 DCD and the supporting TRs have addressed how access controls are incorporated into the design of the PMS.

Regulations in 10 CFR 52.47(a)(9) require applications for light-water-cooled NPPs to evaluate the standard plant design against the NUREG-0800 revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the NUREG-0800 acceptance criteria. NUREG-0800 Section 7.9 provides the design considerations for data communications systems, including criteria for performance and reliability considerations. The staff evaluated the data communications systems in the AP1000 DCD against the guidance provided in NUREG-0800 Section 7.9, which states that digital computer timing should be consistent with the limiting response times and characteristics of the computer hardware, software, and data communications systems. The staff found the applicant's commitment to modify the acceptance criteria in Item 11d) of ITAAC Table 2.5.2-8 in the AP1000 DCD to state "Performance of system tests and the documentation of system test results, including a response time test will be performed under maximum CPU loading to demonstrate the PMS can fulfill its response time criteria" is acceptable to address the criteria in NUREG-0800 Section 7.9 and, therefore, meets the requirements of 10 CFR 52.47(a)(9).

GDC 21 requires the protection system to be designed for high functional reliability and inservice testability, commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to ensure that: (1) no single failure results in loss of the protection function; and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy, unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic tests of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy. Based on the staff's conclusions of the PMS Computer Security Plan, as discussed in Section 7.9.5 of this report, the staff finds that the applicant sufficiently addressed the criteria in Regulatory Positions C.2.1 through C.2.5 of RG 1.152, Revision 2 to meet the requirements of 10 CFR Part 50, Appendix A, GDC 21; Criterion III of Appendix B to 10 CFR Part 50; and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9, as it relates to security and reliability of the PMS application and CIM. As such, the staff concludes that the AP1000 design meets the requirements of 10 CFR Part 50, Appendix A, GDC 21; Criterion III of Appendix B to 10 CFR Part 50; and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9; as it relates to security and reliability of the PMS application and CIM.

GDC 24 requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Based on the review of the interfaces between the PMS and the PLS, the staff concludes that this requirement has been fully satisfied, as documented in Section 7.9.3. The staff finds that the information presented in the AP1000 DCD and the supporting TRs have sufficiently demonstrated how independence is achieved for each of the five cases of safety and nonsafety communications.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

APPENDIX 7.A: EVALUATION OF APP-GW-GLR-137, REVISION 0, "BASES OF DIGITAL OVERPOWER AND OVERTEMPERATURE DELTA-T (OPΔT/OTΔT) REACTOR TRIPS"

7.A.1 Introduction

This safety evaluation addresses changes made from Revision 15 to Revision 19 of the AP1000 DCD regarding a change in methodology for the thermal overtemperature delta-temperature (OTΔT) and thermal overpower delta-temperature (OPΔT) reactor trip design bases. Revision 15 of the DCD references WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions." This is the previously approved topical report for the analog calculation of the reactor trip functions. Revision 19 also references WCAP-8745-P-A; however, Revision 19 includes a change from an analog-based OTΔT and OPΔT design to a digital-based design with a different calculational methodology for the trip function. The basis of the setpoint calculations is unchanged from that presented in WCAP-8745-P-A, but the inputs to both margin-to-trip functions have changed. The digital-based core power indication described in APP-GW-GLR-137, Revision 1, proposes the use of density at the reactor core inlet and the enthalpy difference between the exit and inlet of the core, referred to as the "ΔT power signal," to provide a more accurate measurement of core power. The new TR also claims the setpoint functions have been simplified.

7.A.2 Evaluation

7.A.2.1 Background

RAI-SRP16-CTSB-42 requested that the applicant either submit a previously approved reference supporting the changes to the OTΔT and OPΔT trip functions or submit a reference that supports the changes. The RAI also requested that the applicant comply with generic letter (GL) 88-16, "Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits," to include the appropriate and approved methodology regarding the revised trip functions in Technical Specification (TS) Section 3.3.1-1 and bases, which are addressed in Chapter 16 of this report.

The response to RAI-SRP16-CTSB-42 led to the generation of Open Item OI-SRP16-CTSB-42, since the response did not fully address the staff's request. SER Chapter 16, which is based on changes to the DCD between Revision 15 and Revision 17, references the open item.

The initial response to Open Item OI-SRP16-CTSB-42 was submitted to address staff concerns. A reference to APP-GW-GLR-137, Revision 1, was given to provide the information requested by the staff. Changes to the DCD for Revision 17 were also provided to maintain consistency between the submitted TR and the following DCD sections:

- Section 7.2.1.1.3
- Section 7.2.4
- TS Table 3.3.1, Notes 1 and 2

The staff submitted RAIs based on the review of APP-GW-GLR-137. After reviewing the responses, the staff found the information given in APP-GW-GLR-137 to be acceptable in support of the digital-based OPΔT and OTΔT reactor trip function methodology.

7.A.2.2 Proposed Change

The OPΔT and OTΔT trips are used to protect the specified acceptable fuel design limits (SAFDLs) so as to maintain the fuel in a geometry amenable to cooling. The design basis of the OTΔT trip is to prevent a departure from nucleate boiling (DNB) on all fuel surfaces, while the design basis of the OPΔT trip is to prevent excessive fuel centerline temperatures for all fuel rods.

In the analog technology of the OPΔT trip setpoint, as described in WCAP-8745-P-A, the setpoint is calculated as a function of the average coolant temperature (T_{AVG}) and a core power reduction term related to adverse axial offset. The OTΔT setpoint has the same inputs as the OPΔT trip setpoint with the addition of pressurizer pressure. The setpoints use T_{AVG}, axial offset, and pressurizer pressure (only for OTΔT) as inputs to a dynamically compensated function to determine the percent of rated thermal power (RTP) at which the reactor should trip. The analog signals are converted to ΔT signals by adjusting gains. The basis for the determination of both ΔT setpoints is derived from thermal design limits as explained in WCAP-8745-P-A.

The change from the analog technology to the digital technology is in both the reactor trip function and the RTP measurement. The analog method uses ΔT as a measure of core power, while the digital method uses actual core power. In the digital method, core power is determined by calculating an enthalpy difference between the core inlet and outlet. The inputs from the respective protection system divisions used to calculate the enthalpy terms are T_H for the outlet, T_C for the inlet, and pressurizer pressure, which is used in both terms. Core average temperature is eliminated as the major functional variable in the digital-based function.

In both the analog and digital technology, the OPΔT trip setpoint uses only a preset bias based on a percentage of RTP, which is based on pre-determined thermal limits, as discussed in WCAP-8745-P-A. This term is compared to core thermal power and then dynamically compensated to obtain a margin to trip signal.

In the proposed change to digital technology, the OTΔT trip setpoint directly translates DNB thermal design limits, which give core inlet temperature as a function of RTP for various pressurizer pressures, into inputs for the setpoint calculation. Core inlet temperature and pressurizer pressure information are linearly interpolated from a table that provides the corresponding RTP to serve as the appropriate setpoint. This setpoint is compared to the core thermal power calculation and then dynamically compensated to obtain a margin to trip signal.

7.A.2.3 Regulatory Basis

10 CFR Part 50, Appendix A, GDC 10, “Reactor Design,” states: “The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.” GDC 10, therefore, applies directly to the design of the OTΔT and OPΔT reactor trips since they are part of the reactor protection system.

Furthermore, GL 88-16 was issued to allow licensees to update all applicable cycle-specific limits without formal review by the NRC. These cycle-specific limits are now located in the core operating limits report (COLR) and are referenced throughout TS. The methodologies by which the cycle-specific limits are updated undergo a formal review by the NRC and are referenced in Section 3.3.1 and bases of the TS.

Since the applicant has revised the method by which the OPΔT and OTΔT reactor trip functions are calculated in the digital methodology, as discussed in APP-GW-GLR-137, the new methodology must be reviewed and approved by the staff. Furthermore, once the review is completed and approved, TS 3.3.1 and bases should be updated to reflect the methodologies that serve as the basis for determining the limits given in the COLR.

7.A.2.4 Evaluation

As previously discussed, all methodologies used to update limits given in the COLR must be reviewed and approved by the staff. The change in calculational methodology of the OPΔT and OTΔT reactor trip setpoints is given in APP-GW-GLR-137. The staff reviewed the document and submitted RAIs to better understand how the trip function calculations have changed and to ensure that SAFDLs will not be exceeded.

A review of the RAI responses in support of Open Item OI-SRP16-CTSB-42 closure is provided in the following discussion.

- Question 1 asks how the single failure criterion of 10 CFR 50.55a(h) is met when an individual resistance thermowell detector (RTD) in one of four divisions votes for a trip due to an approach to a saturation condition. In the response, it is stated that the single failure criteria is not impacted by an RTD in a saturated condition since the 2oo4 voting logic by division is unaffected. When a single RTD approaches saturation, it is removed from the average hot leg temperature calculation for that division. Further, it was stated that when two RTDs approach saturation, a trip vote occurs in the affected division. Based on the applicant's response, it was determined that 10 CFR 50.55a(h) has not been violated and Question 1 is resolved.

  Question 2 discusses the accuracy of the ∆T power signal and asks how the bias applied to the Thot-local signal leads to approximation of the mixed mean hot leg temperature. The concern is that the correction factor applied to the individual hot leg RTDs, which are used to calculate the average hot leg temperature, might lead to an inaccurate calculation of the ΔT power signal. The response states that the ΔT power signal is frequently calibrated, as required by TS;, consequently there is reasonable assurance that the streaming bias applied to the Thot-local signals will not affect the calculation of the ΔT power signal used in the margin to trip calculation. Based on the applicant's indication that the ΔT power signal is continuously monitored and validated, Question 2 is resolved and the applicant indicated that the TR will be updated for clarification. These changes were incorporated in a subsequent revision to the TR.

- Question 3 is related to the discussion of the redundant sensor algorithm and asks if the discussion is included in Chapter 7 of the DCD. The applicant stated that APP-GW-GLR-137, which includes the discussion, is referenced in Chapter 7 of the DCD. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Question 4 is concerned with how the weighted averaging of the $T_{hot\text{-}local}$ signals is performed for T_H determination. Specifically, the concern is that automatic adjustment of weighting factors might violate 10 CFR Part 50, Appendix B, Criterion III, "Design

Control." The response states that weighting factors are not changed. If one of the $T_{hot\text{-}local}$ signals is dropped from the determination of T_H, due to an approach to saturation condition, then the same weighted average is performed, only with two weighting factors instead of three. Based on the applicant's response, it is clear that the weighting factors are not adjusted and there is no violation of 10 CFR Part 50, Appendix B, Criterion III. The applicant committed to update the TR for clarification. These changes were incorporated in a subsequent revision to the TR.

- There was concern with how a failure of the core inlet temperature signal (TC) would affect the protection system. Question 5 asks how the system responds to a "BAD" quality TC signal. The response states that an alarm is actuated in the MCR to notify operators to take appropriate action. The value of the signal (e.g., failed off-scale low, failed off-scale high or otherwise) will determine whether or not a trip is voted for. Based on the applicant's response, it is clear that a failure of a TC signal will result in appropriate operator action and, therefore, Question 5 is resolved.

- To determine how the margin to trip function interfaces with the protection system bi-stable controller, Question 6 asks how the margin to trip signal feeds into the logic that generates a division trip vote and if the margin to trip signal is used elsewhere. The response states that the margin to trip signal is directly input to the trip bi-stable controller, which looks for a negative value to allow for a trip vote. The signal also goes to the MCR for alarm and display. It is also stated that the margin to trip signal is hardwired into the PLS and that the information is available for use by other systems if needed. Based on clarification of how the margin to trip signal interfaces with the protection system, Question 6 is resolved.

- Question 7 refers to the use of time constants in the TH, TC, and OPΔT and OTΔT margin to trip signal development. Clarification was requested regarding the extra lag term and also how the values of these constants differ from those in the previously approved analog methodology. The response indicates that the extra lag term is dedicated to signal noise filtering and does not affect the shape of the output signal (i.e., it is comparable to the output signal that uses a first order lag term). The TR discusses the possible factors that go into calculation of the net lead and lag constants, many of which are optional, and furthermore are determined in the plant-specific safety analysis of record to verify the adequacy of the protection system. The responses to Questions 9-11 give typical values assumed in performed analog versus digital comparative analyses. Based on the provided clarification, Question 7 is resolved.

- Question 8 refers to the use of the bias coefficient and the conversion factor used in calculating the ΔT power signal. The question asks how the constants are determined and how often they must be adjusted. The response states that the bias coefficient will be adjusted so that the ΔT power signal indicates zero at hot zero power. This action is mandatory as part of the channel calibration required by TS surveillance requirement (SR) 3.3.1.9, which requires calibration every 24 months. The conversion factor is a gain adjustment that is adjusted as necessary in compliance with SR 3.3.1.3, which requires comparing the ΔT power signal to the calorimetric power, similar to the neutron flux power range signal surveillance, every 24 hours. Based on the provided clarification, Question 8 is resolved.

- Questions 9 through 11 ask the applicant to discuss the differences between the analog and digital based dynamic response when a trip occurs. Provided in the response are the assumed time constants used in comparing the analog response to the digital response. The differences are shown in a comparative example. Additional concern was expressed with regard to the Chapter 15 design basis accidents, which credit the OPΔT and OTΔT reactor trips. It was asked if the Chapter 15 accidents were revised to include the revised digital-based reactor trip functions.

  The comparisons between the two trip responses given in response to RAI-TR36-012 show that the trip responses are similar, which provides reasonable assurance that the dynamic compensation terms applied to the digital-based method and the proposed trip functions are appropriate. It was stated that the Chapter 15 design basis accidents were not updated to reflect the digital-based functions since the comparative studies performed confirmed that the digital-based method closely simulates the analog-based method without a loss of safety margin. In Revision 19 of the DCD, the applicant replaced the analysis results in DCD Section 15.4.2 with the re-analysis results described in response to RAI-TR36-012, which were based on the same cases previously analyzed. Various parts of DCD Section 15.4.2 are revised to reflect the revised analysis and results, as shown in Table 15.4-1 and Figures 15.4.2-1 through 15.4.2-15. The revised analysis used the upgraded digital-based OTΔT reactor trip function and NRC-approved methods, including the revised thermal design procedure (RTDP) and the LOFTRAN code. The revised analysis maintains the consequential loss of offsite power occurring 3 seconds after the turbine trip. Because the AP1000 PMS design is such that turbine trip occurs 5 seconds following a reactor trip condition being reached, the loss of offsite power and the resulting RCP coastdown occur 8 seconds after the reactor trip. Since the minimum DNBR occurs immediately after the reactor trip, the loss of offsite power has no effect on the minimum DNBR results. The revised results continue to show that the DNBR does not fall below the design limit DNBR. Therefore, fuel integrity and adequate fuel cooling are maintained. The peak reactor coolant system (RCS) pressure remains less than 110 percent of the design pressure. Therefore, the revised analysis continues to meet the acceptance criteria of NUREG-0800 Section 15.4.2. Questions 9 through 11 are, therefore, resolved.

- Question 12 was asked to resolve a discrepancy between time constants reported in APP-GW-GLR-137 and those shown in Figure 7.2-1 (Sheet 5) of the DCD. An updated figure was provided with the RAI responses for consistency, and Question 12 is therefore resolved. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Question 13 asked the applicant to ensure that the TS and bases are consistent with the information provided in APP-GW-GLR-137. The response states that appropriate changes are being made. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Section 5.0, “References,” of APP-GW-GLR-137 needs to be updated since the currently referenced topical report discusses the methodology used to determine certain uncertainties that factor into the calculation of the OPΔT and OTΔT setpoints. APP-GW-GLR-137 states that a revision to the topical report will be issued at a later date. This change was incorporated in Revision 1 to the TR.

The staff concludes that, based on the information provided in APP-GW-GLR-137 and the responses to RAIs, no significant change was made to the functional output and underlying methodology for the digital based OPΔT and OTΔT margin to trip functions and it is, therefore, concluded that SAFDLs will not be exceeded. Improvements to the measurement of core power for input to the margin to trip calculation were made by using actual core parameters versus using differential temperature. This change addresses a source of previous inaccuracy in the trip functions. To provide further assurance that the ΔT power signal used in the margin to trip calculation is valid, the signal is also compared to the plant calorimetric heat balance routinely performed in SR 3.3.1.3 as given in Chapter 16 of the DCD.

The setpoint calculated in the digital-based methodology for the OTΔT setpoint is based on allowable core power as a function of pressurizer pressure and core inlet temperature instead of allowable ΔT as a function of T_{AVG}. This is a simpler method and allows for direct translation of the appropriate DNB thermal design limits into the OTΔT trip function. The OPΔT setpoint calculation is unchanged in the digital methodology, except for the units. The OPΔT setpoint is manually fixed at a determined power level and only changes as a function of the adverse axial offset.

7.A.3 Conclusion

After reviewing APP-GW-GLR-137, the staff finds the proposed OPΔT and OTΔT reactor trip function calculational methodology to be acceptable and considers Open Item OI-SRP16-CTSB-42 resolved.

8. ELECTRIC POWER SYSTEMS

8.2 Offsite Power Systems

8.2.2 Offsite Circuits within the AP1000 Scope of Design

In the second paragraph of Section 8.2.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the Nuclear Regulatory Commission (NRC) staff stated that the main generator normally provides power to the main alternating current (ac) power system. When the main generator is not available, the generator output breaker is opened and the plant auxiliary power comes from the switchyard by back feeding through the main step-up transformers and the unit auxiliary transformers (UATs). There is also a maintenance source provided through a reserve auxiliary transformer (RAT). The maintenance source is site-specific, and bus transfer to the maintenance source is manual. In Section 8.2.3.4, "Specific Interface Requirements for Supporting Chapter 15 Analyses," of NUREG-1793, the NRC staff stated that the AP1000 design uses no automatic transfers of reactor coolant pump (RCP) buses to alternate power supply. In addition, in Section 8.3.1, "Onsite Power Systems," of NUREG-1793, the NRC stated that the 6.9 kilovolt (kV) buses are provided with access to the maintenance source through normally open circuit breakers connecting the bus to the RAT and that bus transfer to the maintenance source is manual.

In technical report (TR) (TR-79), "Electrical System Design Changes," Revision 1, the applicant added a fast bus transfer scheme, along with an operator-initiated maintenance transfer, to the AP1000 design. This change required installation of an additional RAT to allow complete bus transfer from UATs to RATs.

As a result of the above changes to Tier 2, the corresponding portions of the Tier 1 AP1000 design control document (DCD) Table 2.6.1-3 (untitled) and Figure 2.6.1-1, "Main ac Power System," were affected.

8.2.2.1 Evaluation

The proposed change will allow transfer of 6.9 kV RCP buses from the UATs to the RATs. The applicant stated that the addition of the fast bus transfer scheme will avoid a reactor trip resulting from component failure or spurious actuation of the protective relaying associated with any of the main step-up transformers, UATs, or isophase bus duct, which would cause an RCP trip and a reactor trip. Thus, when either normal or preferred power supply is unavailable because of an electrical fault at the main step-up transformer, UAT, isophase bus duct, or nonsegregated bus duct, fast bus transfer will be initiated to transfer the loads to the RATs. In addition to the above, the applicant has also added operator-initiated, sync-supervised, closed-transition transfers on a bus-by-bus basis to the AP1000 design.

The staff was concerned about a statement in Section 8.3.1.1.1 of the AP1000 DCD, which reads:

> …in the event of a loss of voltage on these buses, the diesel generators are automatically started and connected to the respective buses and in the event of a fast bus transfer, the diesel generator connection to the bus is delayed such that the fast bus transfer is allowed to initiate.

This statement implies that the diesel generator is already running during the fast bus transfer and its connection to the bus is delayed. In request for additional information (RAI) RAI-SRP8.3.1-EEB-02, the NRC requested that the applicant clarify when the diesel generator would start during fast bus transfer. In its response dated July 11, 2008, the applicant revised Section 8.3.1.1.1 of the DCD to clarify the statement as follows:

> In the event where a fast bus transfer initiates but fails to complete, the diesel generator will start on an undervoltage signal, but if a successful residual voltage transfer occurs, the diesel generator will not be connected to the bus, as the successful residual voltage transfer will provide power to the bus prior to the diesel connection time of 2 minutes.

The staff concluded that this revision to the AP1000 DCD satisfies its concern. The staff verified that Revision 19 to the AP1000 DCD includes the foregoing change.

8.2.2.2 Conclusion

The staff has reviewed these changes and concludes that the additions of a RAT and the bus transfer scheme to the AP1000 design provide additional plant availability and enhance the offsite power supply to the safety-related battery chargers, RCPs, and those priority loads provided for defense-in-depth functions.

8.3 Onsite Power System

8.3.1 AC Onsite Power System

8.3.1.1 Electric Circuit Protection

In this section of NUREG-1793, the NRC discussed the ratings and major types of protection systems employed for the AP1000 medium voltage switchgear. In TR-79, the applicant made the changes described in the sections below.

8.3.1.1.1 Rating of 6.9-kV Switchgear Buses

In TR-79, the applicant proposed to revise the short-circuit rating of 6.9-kV switchgear buses from the current 500 megavolt amps (MVA) (40 kilo-amperes (kA)) to 63 kA. The Tier 2 portions of the AP1000 DCD affected in the electrical area include Figure 8.3.1-1, “AC Power Station One Line Diagram,” and Table 8.3.1-3, “Component Data – Main Power System.”

8.3.1.1.1.1 Evaluation

Originally, the AP1000 DCD included the short-circuit rating of the 6.9-kV switchgear buses as 500 MVA (approximately 40 kA). The applicant proposed to revise the short-circuit rating from the current 40 kA to 63 kA. The applicant stated that the value change from 40 kA to 63 kA is based on revised short-circuit calculations that demonstrate that 63 kA is bounding, given the UAT/RAT size and the expected largest motor size driving the allowed impedance of the UAT/RAT transformers. In RAI-SRP8.3-EEB-01, the staff asked the applicant to justify this change in terms of the reason the interrupting rating changed from 40 kA to 63 kA and whether the proposed change affects the onsite distribution system analysis.

In its response dated October 17, 2008, the applicant stated that the onsite distribution system analysis supports the described engineering values. The value change from 40 kA to 63 kA is based on a computation of short-circuit current from an infinite source upstream of a UAT/RAT, neglecting the minimal contribution between the X-Y secondary windings of the transformer. In addition, the applicant considered a conservative assumption of a 100-percent motor load on a 100-percent loaded transformer winding using a 6.5 multiplier for motor short-circuit contribution. This value was computed while establishing a transformer impedance low enough to allow for starting the single largest motor. This computation demonstrates that 40 kA is inadequate and that 63 kA is bounding given the UAT/RAT size and the expected largest motor size driving the allowed impedance of the UATs/RATs.

8.3.1.1.1.2 Conclusion

The NRC reviewed this change and concludes that the proposed rating of 63 kA is acceptable because, in the current design, the starting current value of the largest motor is 58 kA, which is well in excess of 43 kA for the 6.9-kV switchgear bus rating. Therefore, a 63-kA rating for the 6.9-kV switchgear bus bounds the largest motor. Based on this discussion, the issue is resolved.

8.3.1.1.2 Air Cooled Chillers

In TR-79, the applicant proposed to revise AP1000 DCD Figure 8.3.1-1 to show the air-cooled chillers VWS-MS-02 and VWS-MS-03 being fed from the 6.9-kV buses ES-1 and ES-2, respectively.

The Tier 2 portions of the AP1000 DCD affected include Figure 8.3.1-1.

8.3.1.1.2.1 Evaluation

Figure 8.3.1-1 of the AP1000 DCD did not show the air-cooled chillers VWS-MS-02 and VWS-MS-03 being fed from the 6.9-kV buses ES-1 and ES-2, respectively. These loads were previously connected at the 480-volt (V) level. The applicant revised the DCD to reflect the connection of these loads directly to the 6.9-kV buses.

8.3.1.1.2.2 Conclusion

The staff reviewed this change and concludes that these loads were erroneously shown connected at the 480-V level and that the connection of the air chillers to the 6.9-kV buses is consistent with the design for this size of motor load. Therefore, the proposed change is acceptable.

8.3.1.1.3 Raw Water Feeder Breaker Change

In TR-79, the applicant proposed to revise AP1000 DCD Figure 8.3.1-1 to show the three raw water pumps and their auxiliaries. The Tier 2 portions of the AP1000 DCD affected include Figure 8.3.1-1.

8.3.1.1.3.1 Evaluation

AP1000 DCD Figure 8.3.1-1 in Revision 15 showed that three feeders for three raw water pumps were fed directly from 6.9-kV buses and their auxiliaries were powered from 480-V buses. The applicant has proposed to have each of these three feeders support the respective raw water pump and the associated auxiliaries of that pump. The proposed change allows a single feeder to the raw water pump house and distributes power within that structure.

8.3.1.1.3.2 Conclusion

The staff reviewed this change and concludes that it has no impact on the safety systems; therefore, it is acceptable.

8.3.1.2 Standby Diesel Generators

Section 8.3.1.2, “Standby Diesel Generators,” includes the staff’s review of generator exciter and voltage regulator systems as well as power sources.

8.3.1.2.1 Generator Exciter and Voltage Regulator Systems

In the sixth paragraph of Section 8.3.1.2 of NUREG-1793, the staff stated that the generator exciter and voltage regulator systems are capable of providing full voltage control during operating conditions, including postulated fault conditions.

In TR-79, the applicant proposed to delete the word “static” from the description of the exciter type so that a more readily available design may be used.

In addition, the applicant revised the nominal power ratings of various pieces of equipment and diesel generator loading in Table 8.3.1-2, “Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads,” of the AP1000 DCD.

8.3.1.2.1.1 Evaluation

There are no regulatory criteria that require a specific diesel generator exciter type. Revision 15 of the AP1000 DCD specified that each onsite diesel generator has a “static” exciter. NUREG-1793 did not identify what type of exciter was used for standby diesel generators. The staff concludes that deleting the word “static” from the description of the exciter type will provide applicant flexibility in choosing what equipment is to be procured and there is no impact on performance requirements.

In Revisions 16 and 17 of the AP1000 DCD, the applicant revised the rating and operating load sizes of various pieces of equipment and updated the affected diesel loading tables. The staff reviewed the revised loads in Tables 8.3.1-1 and 8.3.1-2 and found that the total loads exceed the rating of diesel generators. The portions of the AP1000 DCD affected include Tables 8.3.1-1 and 8.3.1-2. In RAI-SRP8.3.1-EEB-01, the staff noted that the sum of the total loads (automatic and manual) listed in each revised table exceeds the continuous rating of each diesel generator and requested the applicant to justify why it is acceptable to exceed the continuous rating of the diesel generator at this stage of the design. Also, the applicant was requested to describe provisions included in the design that will prevent overloading of the diesel generators when manual loads are powered from the diesel generator.

In its response dated June 23, 2009, the applicant stated that onsite standby diesel generators have a nominal rating of 4000 kilowatt (kW); however, the units will accept loads up to the overload ratings of the diesel generators for the period of time specified for those ratings. The intent of the diesel generator loading is to accept all automatic loads followed by other loads to be manually added under operator control. In addition, the applicant stated that the abnormal operating procedures include diesel generator load management details to be followed after the automatic load sequencing. These procedures will identify that additional loads can be manually loaded at the operator's option. The operator will assess plant conditions and available diesel generator capacity to determine if these additional components should be started. The operator has main control room indication of the current power demand on each of the diesel generators upon which to base his decision. The staff finds the response to the RAI to be acceptable.

8.3.1.2.1.2 Conclusion

The staff concludes that removing the word "static" from the type of exciter would provide the applicant more flexibility in choosing other excitation systems; therefore, the proposed change is acceptable. The staff also finds the changes made to Tables 8.3.1-1 and 8.3.1-2 for auto-connected loads to be acceptable because: 1) the total auto-connected loads (2706 kW, 3126 kW) on each diesel generator is still within the continuous rating of 4000 kW; and 2) the procedures that will prevent overloading of the diesel generators when manual loads are powered from the diesel generator are included.

8.3.1.2.2 Power Sources

In the third paragraph of Section 8.3.1.2, "Standby Diesel Generators" of NUREG-1793, the staff stated that during plant startup, shutdown, and maintenance, the generator breaker is opened. Under this condition, the preferred power supply system provides the main ac power from the high-voltage switchyard through the main step-up transformers and two UATs. Each UAT supplies power to about 50 percent of the plant loads. The UATs have two identically rated 6.9 kV secondary windings.

In TR-114, "AP1000 Auxiliary Building Boiler Sizing and Design," the applicant added a third two-winding UAT sized to accommodate the electric auxiliary steam boiler and site-specific loads. In addition, in Revision 17 to the AP1000 DCD, the applicant revised Section 8.3.1.1.1, "Onsite AC Power System," to describe the neutral overcurrent protection for the RATs.

As a result of the above changes to Tier 2, the portions of the Tier 1 AP1000 DCD affected include Figure 2.6.1-1, "Main AC Power System," and Tables 2.6.1-3 and 2.6.1-5.

8.3.1.2.2.1 Evaluation

The applicant has proposed a design change from a diesel-fired auxiliary steam boiler to an electric auxiliary steam boiler for the AP1000 to alleviate issues in current plants related to operational problems caused by fuel fouling in diesel-fired boilers in standby service. This change reduces the size requirement on the auxiliary boiler by over 50 percent (from approximately 70 megawatts (MW) to 25 MW). As a result, the applicant added a third two-winding UAT sized to accommodate the electric boiler and site-specific loads. The third UAT would be located outside the turbine building in the transformer area. A 25-MW electric boiler would be installed in the boiler room of the turbine building along with its associated switchgear ES7, load center, and motor control center. This design change would not have any impact on the safety systems.

The staff was concerned that the applicant did not provide any neutral overcurrent protection for the RATs. In RAI-SRP8.2-EEB-03, dated March 13, 2008, the staff asked the applicant to justify its failure to provide neutral overcurrent protection for the RATs. In its response dated April 22, 2008, the applicant stated that it had inadvertently omitted the neutral overcurrent protection from page 8.3-2 of the AP1000 DCD. The applicant committed to modify DCD Section 8.3.1.1.1 to show neutral overcurrent protection for the RATs. The staff finds the above design to be consistent with the recommendations of Institute of Electrical and Electronics Engineers (IEEE) Standard 666, "IEEE Design Guide for Electric Power Service Systems for Generating Systems."

The staff confirmed that Revision 19 to the AP1000 DCD includes the neutral overcurrent protection for the RATs.

8.3.1.2.2.2 Conclusion

The staff reviewed this change and concludes that the addition of the third UAT and its associated switchgear ES7 has no impact on the safety systems and, therefore, the proposed change is acceptable. In addition, the staff finds the inclusion of the overcurrent protection for the RATs acceptable because it is consistent with expected engineering practice.

8.3.1.3 Ancillary AC Diesel Generators

In the first paragraph of Section 8.3.1.2 of NUREG-1793, the staff stated that the applicant has included two ancillary diesel generators located in the annex building to provide power to meet the post-72-hour power requirements following an extended loss of offsite power sources. Each ancillary diesel generator output is connected to a distribution panel.

In TR-79, the applicant proposed to revise Figure 8.3.1-3 of AP1000 DCD Tier 2 to reflect a four-wire 100 amp distribution panel from a three-wire 50 amp distribution panel, and a 100 amp breaker for both the diesel generator to the bus and for the test load tie to the bus from the 50 amp breaker.

8.3.1.3.1 Evaluation

There are no regulatory requirements concerning the number or rating of wires to or from a nonsafety-related ancillary diesel generator. The staff reviewed this change to determine whether it would adversely affect the design. Currently, AP1000 DCD Figure 8.3.1-3 shows the size of the ancillary diesel generator distribution panels as 50 amp with an incoming breaker of 30 amp from the generator. The applicant proposed to revise Figure 8.3.1-3 of the DCD to reflect a four-wire 100 amp distribution panel from a three-wire 50 amp distribution panel, and a 100 amp breaker for both the diesel generator to the bus and for the test load tie to the bus from the 50 amp breaker. The applicant stated that since the full load current of the generator is 53 amps, the main breaker of the distribution panel should be sized at the full capacity of the generator at a minimum. The diesel generator test load will also be changed to 100 amp to allow for this generator to be tested at full capacity. The applicant made its selection based on the 480 volts alternating current (Vac) standard-sized distribution panels available in the industry. In addition, to facilitate the use of this source as a feed to 277 Vac lighting circuits, these panels would be changed from three-wire system to four-wire system.

8.3.1.3.2 Conclusion

The staff reviewed these changes and concludes that the original rating of the bus and breakers of the panel was undersized and that the proposed revised rating of the diesel generator distribution panel is adequate because it exceeds the full load current of the diesel generator. Therefore, the proposed change is acceptable.

8.3.2 Direct Current Power and Uninterruptible Power Systems

8.3.2.1.1 Class 1E dc Distribution

In Section 8.3.2.1.1 of NUREG-1793, the staff stated that the Class 1E direct current (dc) power system consists of four independent 125 V Class 1E dc safety system divisions (Divisions A, B, C, and D). Divisions A and D are each comprised of one battery bank, one switchboard, and one battery charger. Divisions B and C are each comprised of two battery banks, two switchboards, and two battery chargers; however, in Revision 17 of the AP1000 DCD, the applicant changed the system voltage for the operation of Class 1E dc loads from 125 volts direct current (Vdc) to 250 Vdc.

The portions of the AP1000 DCD affected include pages 8.1-2, 8.1-3, 8.3-10, 8.3-14 through 8.3-17, 8.3-22 through 8.3-26, and Tables 8.3.2-1 through 8.3.2-7. The portions of the Tier 1 AP1000 DCD affected include Section 2.6.3, "Class 1E dc and Uninterruptable Power Supply System," Table 2.6.3-1 (untitled), Table 2.6.3-3, "Inspections, Tests, Analyses and Acceptance Criteria," and Table 2.6.3-4 (untitled).

As part of the staff's review of the proposed changes to the Class 1E dc system, the staff issued several RAIs. The staff's evaluation of the applicant's responses to the RAIs is as follows.

8.3.2.1.1.1 Evaluation

In RAI-SRP8.3.2-EEB-01, the staff requested that the applicant provide a discussion as to how this voltage change would impact motors, cables, protective devices, switchboard, and other equipment, as applicable. Also, the applicant was asked to describe how motor sizing and cable sizing would still be compatible with valve loads.

In its response dated May 7, 2009, the applicant stated that dc motor-operated valve motors would draw less current to accomplish the required power. Cable sizes would be reduced considerably. As the current drawn by the same size (kW) motor is halved and the total voltage drop allowed is doubled, the cable sizes would be reduced accordingly. Electrical distribution equipment would require, nominally, one half the current rating. The staff agrees with the applicant that by increasing the system voltage the current would be reduced, the cable sizes would be reduced, and the electrical distribution equipment would require less current. Therefore, the staff finds applicant's response to be acceptable and finds this issue resolved.

In RAI-SRP8.3.2-EEB-02, the staff noted that AP1000 DCD Section 8.3.2.1 indicates that the operating voltage range of the Class 1E batteries is 210 to 280 Vdc and the maximum equalizing charge voltage for the Class 1E batteries is 280 Vdc. The applicant was asked to confirm that the connected dc equipment is designed to operate up to the maximum voltage 280 Vdc.

In its response dated May 7, 2009, the applicant stated that all connected equipment design specifications would include the new voltage limit requirements. Based on the above, the staff finds that since specification of each piece of equipment would include the revised voltage specification, the applicant's response is acceptable and this issue is resolved.

In RAI-SRP8.3.2-EEB-03, the staff requested that the applicant provide the load profiles (duty cycle) from one minute to 24/72 hours for each of the 24-hour and the 72-hour Class 1E 250 Vdc batteries. The applicant was asked to discuss battery margins (aging margin, design margin, temperature correction factor, margin associated with float current for 100 percent state of charge) and the expected service life of these batteries.

In its response dated June 23, 2009, the applicant stated that for battery aging margin, a factor of 25 percent would be used for a 20 year qualified battery. Temperature correction would be based on minimum temperature of 16° Celsius (C) (60° F). With regard to float current margin, the applicant stated that this margin is described as a consideration for quick turnaround to service after discharge. Since the electrical design described in the AP1000 DCD utilizes a spare battery that can replace any safety-related battery, there is no immediate need to replace the discharged battery. The replacement interval/service life of the batteries will be in accordance with the testing program replacement requirements. Replacement intervals will be based on degraded performance in accordance with the required test program. However, the applicant did not provide the load profiles for 24-hour and 72-hour batteries as requested by the staff. The staff identified this as Open Item OI-SRP8.3.2-EEB-03 in the safety evaluation report (SER) with open items.

Subsequently, in a letter dated May 10, 2010, the applicant stated that the nominal loads on the batteries are identified in AP1000 DCD Tables 8.3.2-1, 8.3.2-2, 8.3.2-3, and 8.2.3-4 and that the design is based on intelligent assumptions on the loads. Also, as part of the response to the above open item, the applicant provided document, APP-IDS-EOC-001, Revision 0, “Class 1E 250 V DC Battery Sizing, Charger Sizing and Available Short Circuit Current,” for the staff’s review to assess the adequacy of the 24-hour and 72-hour batteries. The staff reviewed the applicant’s onsite documentation which included the load profiles for loss of offsite power and loss-of-coolant accident (LOCA) from one minute to 24/72 hours for each of the 24-hour and the 72-hour Class 1E 250 Vdc batteries in this report. Based on its review, the staff concluded that since the AP1000 Class 1E 250 Vdc batteries are sized in accordance with the recommendations of IEEE Standard 485, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” (which provides methods for defining the dc load and for sizing a battery to supply that load for stationary batteries), there is reasonable assurance that the batteries would be designed to have adequate capacity to meet their respective load profile. In addition, the staff determined that the battery qualification program and the applicable surveillance requirements in accordance with plant technical specifications would ensure that the batteries would envelop their designed load profiles throughout their designed life. On this basis, the staff considers Open Item OI-SRP8.3.2-EEB-03 resolved.

In RAI-SRP8.3.2-EEB-04, the staff requested that the applicant describe how the 24-hour and the 72-hour 250 Vdc batteries would be qualified for service life: If safety-related batteries would be qualified using the recommendations of IEEE Standard 535, “Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations,” it is not clear how the standard applies since the standard was written under the assumption of an 8-hour duty cycle. Since AP1000 design duty cycles are significantly longer than 8-hour duty cycle and IEEE Standard 535 does not apply to duty cycles longer than 8 hours, the applicant was asked to describe how these batteries would be qualified for extended duty cycles of 24 hours and 72 hours. The applicant was also asked to discuss the failure mode(s) for both the 24-hour and 72-hour duty cycle batteries.

In its response dated May 7, 2009, the applicant stated that it intends to qualify the AP1000 safety-related batteries for 24-hour and 72-hour duty cycles through the implementation of industry standards IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"; IEEE Standard 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"; and IEEE Standard 535-1986 as they apply to the equipment.

The qualification process for the AP1000 24-hour and 72-hour duty cycle batteries would be outlined in a test plan. Qualification of the batteries would be accomplished by type testing of both duty cycle designs to the AP1000 service conditions associated with their projected service life. In the qualification process, the batteries would be subjected to aging (thermal, wear/operational), abnormal environmental, and seismic conditions. There are no radiation and normal vibration conditions associated with the mounting locations of the batteries. Aging under normal and abnormal service conditions would be performed to degrade batteries to their end-of-life such that the safety function after the design basis event (DBE) (seismic) would be verified.

The aging conditions would include both electrical (chemical) cycling and thermal accelerated aging. Electrical (chemical) cycling would be performed in compliance with IEEE Standard 323. The proposed electrical (chemical) cycling is in line with Section 8.2.2 (6) of IEEE Standard 535-1986 for cases when the service conditions are more severe than those specified in the standard. The electrical (chemical) cycling of the batteries is based on the AP1000 maintenance/surveillance requirements with no less than 10 percent margin. During the testing process, the service and performance tests would be performed in conjunction with the thermal accelerated aging test of the batteries to place the batteries in an end-of-life condition. Upon completion of the battery aging, abnormal environmental testing to the AP1000 mild environment abnormal conditions would be performed. Following the abnormal environmental testing, seismic testing and a hard rock high frequency screening test would be performed.

At the completion of seismic testing, a post-seismic battery service test would be performed. The service test is used to demonstrate equipment functionality during and after the DBE (seismic), which is a requirement in accordance with IEEE Standard 344 and IEEE Standard 323. This is different from IEEE Standard 535, which only requires a performance test to be performed. In the process of performing the qualification testing of the AP1000 batteries, the program would identify any failure mechanisms that may surface during the projected service life in an AP1000 plant.

In a May 21, 2009 conference call, the staff requested that the applicant provide its step-by-step, detailed qualification test plan showing testing for desired qualified life of the batteries. However, the applicant did not provide its qualification test plan for the batteries prior to issuance of the SER with open items. This issue was tracked as Open Item OI-SRP8.3.2-EEB-04 in the SER with open items.

Subsequently, in a letter dated March 2, 2010, as part of the response to the above open item, the applicant provided document, EQ-TP-59-APP (APP-DB01-VPH-001), Revision 0, "AP1000 Test Plan for Safety Related 250 Vdc Batteries," for the staff's review. The staff reviewed the applicant's onsite documentation supporting the qualification methodology for the 24-hour and 72-hour extended duty cycle batteries. The applicant provided detailed steps that would be followed to qualify the batteries as requested by the staff. The qualification would be based on the requirements of IEEE Standard 323-1974, IEEE Standard 344-1987, and IEEE Standard 535-1986. Qualification of the Class 1E batteries would be performed by testing. Due to the difference in duty cycle, the test sequence would be performed on two groups of test cells. One group would be cycled and tested to the 24-hour duty cycle for AP1000 and the other group would be cycled and tested to the 72-hour duty cycle for AP1000. The test plan includes a series of modified performance tests at two year intervals that envelop the load profile of a service test throughout the installed 20-year life of the batteries.

In a public meeting held on May 27, 2010, to discuss the AP1000 Chapter 8 open items, the staff informed the applicant that its qualification test plan for the batteries is reasonable, but that the applicant must capture the qualification test plan as part of its licensing basis.

In a letter dated June 18, 2010, the applicant stated that it would revise its DCD to include the battery qualification test program. The applicant included the proposed revised Section 8.3.2.1.4, "Description," of the DCD as part of its response as follows:

> The qualification test program for AP1000 24-hour and 72-hour class 1E batteries meets or exceeds the requirements of IEEE Std 323, IEEE Std 344 and IEEE Std 535 including required and recommended margins and is in regulatory compliance with RGs 1.89, 1.100 and 1.158. The test program requires that the battery be subjected to accelerated thermal aging and discharge cycling (wear aging) in accordance with Institute of Electrical and Electronics Engineers (IEEE) Std 323 and IEEE Std 535 over its qualified life objective followed by the DBE seismic event performed in accordance with IEEE Std 344. In addition, following the aging process, the test specimens shall be subjected to environmental testing to verify the equipment's ability to operate in postulated abnormal environmental conditions during plant operation. Discharge cycling will be performed as a potential aging mechanism prior to seismic testing using Type 3 modified performance test method in accordance with IEEE Std 450-2002 at intervals representative of the AP1000 surveillance test requirements of the batteries with 10 percent margin in the number of discharge cycles which establishes margin for the expected life of the battery. Thus, magnitude / duration (modified performance test versus service and performance tests) and test interval envelop the AP1000 and industry cycling requirements. If new battery failure modes are detected during the qualification testing, these failure modes will be evaluated for any potential changes to the technical specification's surveillance requirements and revision to maintenance procedures required to ensure identification of degradation prior to reaching those failure modes during plant operation. Following the qualification process, a report that uniquely describes step-by-step the tests performed and results and addressing any deficiencies and repairs including photographs, drawings, and other materials will be maintained for records.

Based on the above, the staff concluded the applicant's test plan provided in EQ-TP-59-APP (APP-DB01-VPH-001) satisfies the recommendations of IEEE Standard 323-1974, IEEE Standard 344-1987, and IEEE Standard 535-1986 and provides reasonable assurance that its batteries and racks will perform their required functions throughout their qualified life. Therefore, Open Item OI-SRP8.3.2-EEB-04 is resolved subject to the verification that the AP1000 DCD is updated to include the revised paragraph. The staff confirmed that Revision 19 to the AP1000 DCD includes the revised paragraph.

In RAI-SRP8.3.2-EEB-05, the staff stated that in order to assess the adequacy of the dc power systems, it needed the results of 250 Vdc battery and battery charger sizing calculations, battery terminal voltage calculations, short circuit calculations, voltage drop calculations, and the associated assumptions used.

In its response dated May 7, 2009, the applicant stated that the results of 250 Vdc battery and battery charger sizing calculations, battery terminal voltage calculations, short circuit calculations, and voltage drop calculations would be available during a design review stage. This was tracked as Open Item OI-SRP8.3.2-EEB-05 in the SER with open items.

Subsequently, in a letter dated April 21, 2010, as part of the response to Open Item OI-SRP8.3.2-EEB-05, the applicant provided document, APP-IDS-EOC-001, for the staff's review. The document includes information to assess the adequacy of the battery banks and chargers for use in the Class 1E dc and uninterruptable power supply (UPS) for the AP1000 plant. The staff reviewed the applicant's onsite documentation that describes applicant's methodology for sizing batteries, chargers, and the available short circuit current from these sources. The staff verified that AP1000 Class 1E batteries are sized in accordance with the recommendations of IEEE Standard 485, and the battery chargers are sized in accordance with the recommendations of IEEE Standard 946, "IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations." These standards provide guidance for sizing batteries and battery chargers. During its review of the document, the staff noticed that the required capacity of an IDSC-DB Division C, 72-hour battery is 2430 AH, while the batteries are rated for only 2400 AH in accordance with the AP1000 DCD. In a public meeting held on May 27, 2010, the staff asked the applicant to justify the apparent difference in the required capacity of the 72-hour battery per calculation versus the stated capacity of the 72-hour battery listed in Table 8.3.2-5 of the DCD.

In a letter dated June 18, 2010, the applicant stated that the required capacity of the IDSC-DB Division C, 72-hour battery in APP-IDS-EOC-001, Revision 0, would be revised to be made consistent with the AP1000 DCD. The staff finds the applicant's commitment acceptable because the revised calculations will ultimately be reviewed by the staff as part of the inspections, tests, analyses, and acceptance criteria (ITAAC) for the dc system, as described in AP1000 DCD Tier 1, Table 2.6.3-3, "Inspections, Tests, Analyses and Acceptance Criteria."

Based on its review, the staff concluded that the applicant's methodology for sizing batteries and battery chargers using these standards provides reasonable assurance that the batteries and battery chargers will be sized adequately and perform their safety functions as designed. The staff also verified that the dc switchboard rating exceeds the available short circuit current contributions from the batteries, battery chargers and regulating transformers. Further, ITAAC verifying that the batteries, chargers, and distribution systems are adequately designed are identified in AP1000 DCD Tier 1, Table 2.6.3-3. The staff will ultimately verify these ITAAC to ensure that the dc distribution system components, including the batteries and battery chargers, are adequately designed and the as-built design conforms to the approved plant design and applicable regulations. Therefore, Open Item OI-SRP8.3.2-EEB-05 is resolved.

In RAI-SRP8.3.2-EEB-06, the staff noted that AP1000 DCD Section 8.3.2.1.1.1, for 72-hour 250 Vdc batteries states, "Each switchboard connected with a 72-hour battery bank supplies power to an inverter. No load shedding or load management program is needed to maintain power during the required 24-hour safety actuation period." The staff requested that the applicant clarify if manual actions would be necessary to maintain power during the required 72 hours and to describe the loads that would be shed after 24 hours.

In its response dated May 7, 2009, the applicant confirmed that no operator action is necessary during the 72-hour period to maintain the adequacy of either the 24-hour or 72-hour portions of the dc power system. This satisfies the staff's concern and this item is resolved.

In RAI-SRP8.3.2-EEB-08, the staff noted that Figure 2.6.1-1, Tier 1 of the AP1000 DCD shows that the motor control centers that feed the safety-related 250 Vdc battery chargers are fed from 480 V load centers. The applicant was asked to provide a detailed drawing of the 480 V load centers, the motor control center (MCC) that feeds the battery chargers, and the dc MCC showing typical loads powered from these buses. The applicant was asked to describe how the 480 V load centers are protected from degraded voltage and frequency conditions and to provide the following information:

a. The results of an analysis of the onsite power distribution system to demonstrate that adequate voltages at terminals of the battery chargers are optimized for the maximum and minimum voltage variations of the offsite power for events such as a unit trip, LOCA, startup or shutdown.

b. A description of the analytical techniques, methodology, and assumptions used in performing the analyses. Also, provide the results of these analyses for each level of onsite electrical power distribution.

c. Identification of the analytical software (and its version) used for performing these studies and make available to the staff an electronic copy of the electrical distribution system model that forms the basis of the analytical studies.

In its response dated June 23, 2009, the applicant stated that the level of design detail requested would be available following the completion of the design review stage of the system. In general, the staff was seeking understanding of the applicant's approach to assuring consistency in the transfer to the combined license (COL) applicant: 1) analysis, calculations and assumptions made for maintaining adequate voltage regulation at safety-related equipment terminals; 2) the analysis and assumptions used to evaluate acceptable rating for equipment such as circuit breakers; and 3) the studies, acceptance criteria, and assumptions used to determine equipment sizing. This was tracked as Open Item OI-SRP8.3.2-EEB-08 in the SER with open items.

Subsequently, in letters dated February 1, 2010, and May 11, 2010, the applicant stated that to assure consistency in the transfer of design information to a COL applicant, it provides a COL applicant with a configuration-controlled model developed through the use of an Electrical Transient Analysis Program. The program includes the nonsafety-related ac design calculations, including design inputs, assumptions, methodologies, and acceptance criteria used in the development of the sizing basis, settings, load flow, short circuit, and voltage regulation. In addition, the applicant stated that it has performed an analysis of its onsite ac distribution system and that the results of the analysis of the onsite power distribution system demonstrate that adequate voltages at the terminal of the battery chargers are optimized for the maximum and minimum voltage variations of the offsite power for events, such as a unit trip, LOCA, startup or shutdown. The above will ensure that adequate voltages at terminals of the battery chargers are optimized for the maximum and minimum voltage variations of the offsite power to satisfy the requirements of 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 17, "Electric Power Systems," with respect to their capacity and capability to perform their safety function. Although the applicant did not submit this information for staff review, the staff determined this response to be acceptable because ultimately all COL applicants would have to complete testing of the onsite (ac and dc) and offsite power systems, which are fully interconnected, to verify that the non-Class 1E ac power system will have an acceptable design to support the safety-related loads. Therefore, Open Item OI-SRP8.3.2-EEB-08 is resolved.

In RAI-SRP8.3.2-EEB-09, the staff noted that the AP1000 is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The staff is concerned about the transient conditions where a significant voltage spike during islanding could cause high dc voltage conditions on the output side of the battery chargers. Operating experience (see NRC Information Notice (IN) 2006-18, "Significant Loss of Safety-Related Electrical Power at Forsmark, Unit 1, in Sweden" dated August 17, 2006) has shown that the voltage spike due to malfunction of the main generator exciter or during islanding could go as high as 130 percent, which could go undetected by normally-provided relaying and could cause damage to the safety-related equipment or mis-operation. In this regard, the applicant was asked to describe how the protective features of the inverter and the new battery chargers would be coordinated so that any voltage transient would not result in inadvertent loss of the inverters or the batteries.

In a letter dated June 23, 2009, the applicant stated that the battery charger input circuit will conduct power to charge the batteries when ac power is available. The battery charger is specified to return to operation after voltage drifts outside of an acceptable input voltage range. The battery charger is also a qualified isolation device, isolating the battery and the inverter from the nonsafety-related ac system. During the period where the battery charger is not conducting, the battery will carry the load. In addition, the applicant stated that it has considered over-voltage events with the potential to have effects upon plant safety-related equipment as provided under the direction of IN 2006-18. However, the applicant did not provide the details of how to avoid this kind of event in AP1000 design or identification of potential vulnerabilities and actions that could reduce the challenges for the control room operators. This potential event is significant in that it can cause the common mode failure in all four trains and, therefore, could result in the loss of all four trains of safety-related ac and dc power. Transient voltages on the ac input to the battery chargers can result in high dc voltages that could lead to failures of critical electrical and electronic components including electrical inverters unless they are properly protected. During such a voltage transient, the inverter voltage surge protection could trip before actuation of the battery charger protection if the battery charger and inverter dc input voltage protection settings are very close to each other. Therefore, it is necessary that the safety-related battery chargers and inverter trips be coordinated such that the associated inverters do not trip on during voltage transients on the ac distribution system. This was tracked as Open Item OI-SRP8.3.2-EEB-09 in the SER with open items.

Subsequently, in letters dated January 26, 2010, and May 11, 2010, the applicant stated that as part of the component design specification, the battery charger/inverter would be designed specifically with consideration of the Forsmark incident identified in IN 2006-18. Industry evaluations of this incident identify the lack of coordination as a primary causative issue. In addition, the applicant stated that the protective devices will be set so that the battery charger will not trip on the over-voltage resulting from load rejection and will be set low enough to protect the equipment. The inverter dc input protection will be set at least 10 percent higher than the battery charger output dc protection to prevent the inverter tripping before the battery charger.

In a public meeting held on May 27, 2010, to discuss AP1000 Chapter 8 open items, the staff informed the applicant that its response to the above open item was inadequate. The staff stated that the safety-related inverter high dc input voltage trip set point and the associated battery charger high dc output voltage trip set point should be coordinated in both magnitude and time. The staff stated that the applicant should amend the DCD to include its response as modified.

In a letter dated June 18, 2010, the applicant provided a proposed revision to Section 8.3.2.1.4, "Maintenance and Testing," of the AP1000 DCD as part of its response to Open Item OI-SRP8.3.2-EEB-09 as follows:

> The inverter DC input protection will be set at least 10 percent higher than the battery charger trip setpoints to prevent the inverter tripping before the battery charger. The time delay for the inverter high dc input voltage trip will be set higher than the delay time delay for the battery charger to prevent the inverter tripping before the battery charger.

In addition, the applicant stated in its response dated May 11, 2010, that the battery charger function in the AP1000 design is to provide isolation between input ac and the dc system and to provide dc power when ac power is available. The staff noted that Section 8.3.2.2 of the AP1000 DCD states that Class 1E battery chargers and Class 1E voltage-regulating transformers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side. Both have built-in circuit breakers at the input and output sides for protection and isolation. The circuit breakers are coordinated and periodically tested to verify their current-limiting characteristics. In the public meeting held on May 27, 2010, the staff requested the applicant explain how the requirement for periodic testing of the Class 1E battery chargers and Class 1E voltage-regulating transformers used as isolation devices would be satisfied by each COL applicant. The staff requested that the applicant indicate where this requirement would be located so that periodic testing of these devices was performed by each applicant to satisfy the recommendations of IEEE Standard 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," endorsed by Regulatory Guide (RG) 1.75, "Physical Independence of Electric Systems."

In the letter dated June 18, 2010, the applicant stated that a COL information item would be added to the AP1000 DCD to ensure that periodic testing was performed on the battery chargers and the regulating transformers. The applicant included the proposed revised Section 8.3.3, "Combined License Information for Onsite Electric Power," of the DCD as part of its response as follows:

> Combined License applicants referencing the AP1000 certified design will ensure that periodic testing is performed on the battery chargers and voltage regulating transformers.

Based on the above, the staff concluded that the applicant's modified response on battery charger and inverter trip setpoints satisfies the requirements of GDC 17 with respect to the capability of dc systems to perform their safety function. The staff determined that the addition of the above COL information item to the AP1000 DCD will ensure that periodic testing is performed on the battery chargers and the regulating transformers in accordance with the requirements of GDC 18, "Inspection and Testing of Electric Power Systems." The staff confirmed that Revision 19 to the AP1000 DCD includes the foregoing revised paragraph.

In RAI-SRP8.3.2-EEB-10, the staff remarked that Note 8 on Figure 8.3.1-4, "Inside Diesel Generator Building," (Sheets 1 and 2) of the AP1000 DCD indicates that the diesel generators include dc pre-lube oil pumps and keep-warm lube oil heaters and that these loads are not included on Table 8.3.2-2, "250 Vdc Class 1E Division B Battery Nominal Load Requirements," and Table 8.3.2-3, “250 Vdc Class 1E Division C Battery Nominal Load Requirements,” of the DCD. Additionally, the same tables do not include dc power requirements to close and recharge the springs of the circuit breakers, nor do they include the dc power requirements for diesel generator field flashing and starting. The staff asked the applicant to indicate whether the battery sizing will include these loads or provide a reference to where these loads are powered from.

In its response dated May 7, 2009, the applicant stated that the nonsafety-related diesel generator dc loads are not powered from the safety-related batteries. The required diesel generator dc loads will be powered from the nonsafety-related dc system. Nonsafety-related breakers will also have their spring charging motors powered from the nonsafety-related dc system.

The safety-related breakers, reactor trip and RCP trip will receive their control power from the safety-related dc system. Based on the above information, the staff finds this issue resolved.

8.3.2.1.1.2 Conclusion

The staff reviewed the proposed changes to the system voltage for the operation of Class 1E dc loads from 125 Vdc to 250 Vdc and concludes that the proposed changes are acceptable.

8.3.2.3 Non-Class 1E dc and UPS System

In this section of NUREG-1793, the staff stated that the non-Class 1E dc and UPS system consists of the dc electric power supply and distribution equipment that provides dc and uninterruptible ac power to the plant non-Class 1E dc and ac loads that are needed for plant operation and investment protection. Direct current Buses 1, 2, and 3 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E UPS system. Bus 4 supplies large dc motors and other dc power loads, but not inverter loads.

In Revision 17 to the AP1000 DCD, the applicant added another dc subsystem, which includes a battery, a battery charger, and the associated dc distribution equipment, and monitoring and protection devices to serve nonsafety-related loads.

The portions of the AP1000 DCD affected include pages 8.3-18 to 8.3-20, 8.3-24, Table 8.3.2-6 (Sheet 1) and a new Table 8.3.2-6 (Sheet 2). The portions of the Tier 1 AP1000 DCD affected include Table 2.6.1-2 (untitled), Section 2.6.2, “Non-Class 1E dc and Uninterruptible Power Supply System,” and Table 2.6.2-1, “Inspections, Tests, Analyses, and Acceptance Criteria,” Table 2.6.2-2 (untitled), and Figure 2.6.2-1, “Non-Class 1E dc and Uninterruptable Power Supply System” (Sheet 2 of 2).

8.3.2.3.1 Evaluation

The applicant added an additional dc subsystem (EDS5) in the non-Class 1E portion of the dc and UPS system in the AP1000 design, which includes a battery, a battery charger, and the associated dc distribution equipment, and monitoring and protection devices. As a result of the addition of the non-Class 1E dc subsystem EDS5, the large dc motors that were originally powered from EDS4 will now be powered from new EDS5.

8.3.2.3.2 Conclusion

The staff has reviewed the proposed change and concludes that the addition of subsystem EDS5 provides greater flexibility to service the non-Class 1E portion of the dc and UPS system for the AP1000 design. Since the proposed change has no impact on the safety-related systems, the proposed change is acceptable.

8.4 Other Electrical Features and Requirements for Safety

8.4.1 Containment Electrical Penetrations

In the first paragraph of Section 8.4.1 of NUREG-1793, the staff stated that for modular type penetrations (three penetration modules in one nozzle), the applicant has assigned the following:

- one module for low-voltage power
- one module for 120 Vac/125 Vdc control and signal
- one module for instrumentation signal

In TR-79, the applicant deleted the above assigned module separation criteria for cables of varying voltage service levels. In addition, the applicant revised Figure 8.3.1-1 of the AP1000 DCD to correct penetration numbers associated with each RCP.

As a result of the above changes to Tier 2, the portions of the Tier 1 AP1000 DCD affected include Figure 2.6.1-1 (Sheet 1 of 4 and Sheet 3 of 4).

In addition, in Tier 1, Section 2.2.1, “Containment System,” and Table 2.2.1-3, “Inspections, Tests, Analysis, and Acceptance Criteria,” the applicant added a new item “6d” to address environmental qualification requirements for non-Class 1E electrical penetrations to resolve the staff’s concern in RAI-TR93-ICE2-03.

8.4.1.1 Evaluation

The applicant stated that the electrical penetration conductor modules are in penetrations of the same service class. Modules for instrumentation signals will be in instrumentation penetrations; modules for power (e.g., 120/125 V) will be in control penetrations; and modules for low-voltage power will be in low-voltage power penetrations.

In addition to the above, the applicant stated that the penetration numbers shown in Figure 8.3.1-1 of Revision 15 of the AP1000 DCD were incorrect and were revised to reflect the correct penetrations associated with each RCP. The penetration numbers currently shown on the figure are E9, E10, E25, and E26. The correct penetration numbers are P10, P26, P9, and P25 for each of RCP 1B, 2B, 1A, and 2A, respectively.

With regard to qualification requirements for non-Class 1E electrical penetrations, in RAI-TR93-ICE2-03, the staff expressed its concern that non-Class 1E penetrations were not qualified for a harsh environment as were Class 1E penetrations. In its response, the applicant agreed with the staff that Class 1E and non-Class 1E penetrations must be qualified for maintaining their containment integrity to satisfy the requirements of GDC 50, “Containment Design Basis.”

8.4.1.2 Conclusion

The staff has reviewed these changes and concludes that the electrical penetration conductor modules are in penetrations of the same service class. This is consistent with the recommendations of RG 1.75, and is acceptable.

The staff also concludes that including environmental qualification requirements for non-Class 1E electrical penetrations satisfies the requirements of GDC 50 and is acceptable. In addition, the staff finds the proposed change to revise the penetration numbers associated with each RCP to be administrative in nature and acceptable.

8.4.2 Reactor Coolant Pump Breakers

In the first paragraph of Section 8.4.2 of NUREG-1793, the staff stated that the RCPs are powered from the four switchgear buses located in the turbine building. Each bus powers one RCP. Variable speed drives are provided for RCP startup. Two Class 1E circuit breakers connected in series power each RCP. These are the only Class 1E circuit breakers used in the main ac power system for the specific purpose of satisfying the safety-related tripping requirements of these pumps.

In TR-79, the applicant proposed to add input and output isolation breakers to each RCP variable frequency drive (VFD) unit. The proposed change would affect AP1000 DCD Tier 2, Table 8.3.1-3 and page 8.3-53 (untitled electrical drawing).

As a result of the above changes to Tier 2, the portions of the Tier 1 AP1000 DCD affected include Figure 2.6.1-1 (Sheet 1 of 4 and Sheet 3 of 4).

8.4.2.1 Evaluation

The applicant proposed to add input and output isolation breakers to each RCP VFD unit. The applicant stated that the addition of the input and output breakers allows for the VFD unit to be completely removed from service during normal plant operation by using the bypass breaker. Without the addition of these isolation breakers, the RCP pump would need to be offline in order to service the VFD unit.

There are no regulatory requirements concerning the ability to remove VFD units during normal plant operation.

8.4.2.2 Conclusion

The staff has reviewed this change and concludes that the addition of the input and output breakers will provide the applicant flexibility to service the VFD unit without removing the RCP offline and that this change has no impact on the safety systems. Therefore, the proposed change is acceptable.

9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Summary of Technical Information

Section 9.1.1, "New Fuel Storage" of the Westinghouse Electric Company, LLC (Westinghouse or the applicant) AP1000 design control document (DCD), Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant has proposed to make the following changes to Section 9.1.1 of the certified design:

1. New fuel rack design change. The basis for this change is documented in technical report (TR)-44, "New Fuel Storage Rack Structural/Seismic Analysis," APP-GW-GLR-026, Revision 0 of May 2006, and TR-44, Revision 1 of July 2008. TR-44, Revisions 0 and 1 described the design details and design-basis analyses for the new fuel racks. To be consistent with the design of the new fuel racks and the analyses presented in these TR revisions, the applicant proposed several changes throughout Section 9.1.1 of DCD Revision 17.

2. Fuel handling crane change. The applicant proposed to replace references to the fuel handling jib crane for the new-fuel handling crane. The basis for this change is addressed in TR-106, "AP1000 Licensing Design Changes for Mechanical System and Component Design Updates," APP-GW-GLN-106, Revision 1 of September 2007.

3. The applicant proposed conforming changes to DCD Section 9.1.1 related to the new fuel storage rack criticality analysis. The basis for this change is documented in TR-67, "New Fuel Storage Rack Criticality Analysis," APP-GW-GLR-030, May 29, 2006.

9.1.1.2 Evaluation

The staff reviewed all changes identified in the proposed amendment to the AP1000 DCD. The staff did not re-review descriptions and evaluations of the new fuel storage in the AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in the applicant's TRs.

The regulatory basis for AP1000 DCD, Section 9.1.1, is documented in NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," of September 2004. The staff has reviewed the proposed changes to DCD Section 9.1.1 against the applicable acceptance criteria of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Sections 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" and 9.1.2, "New and Spent Fuel Storage." The following evaluations discuss the results of the staff's review.

The specific criterion that applies to the changes evaluated in this section is Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(a)(1)(vii), "Finality of standard design certifications," which concerns contribution to increased standardization of the certification information.

9.1.1.2.1 New Fuel Rack Design Change

9.1.1.2.1.1 Summary of Technical Information

The applicant proposed the following changes for the new fuel racks:

1. In DCD Section 9.1.1.1, deleted "… to supporting grid structures at the top and bottom elevations." and replaced it with "… to a thick base plate at the bottom elevation."

2. In DCD Section 9.1.1.1, deleted "… but may be braced as required to the pit wall structure."

3. In DCD Section 9.1.1.1, added reference to DCD Figure 9.1-1 for rack layout.

4. In DCD Section 9.1.1.1, replaced "new fuel handling crane" with "fuel handling machine."

5. In DCD Section 9.1.1.1, added the sentence "The stress analysis of the new fuel rack satisfies all of the applicable provisions in NRC Regulatory Guide (RG) 1.124, Revision 1 for components design by the linear elastic method."

6. In DCD Section 9.1.1.2, deleted "and laterally supported as required at the rack top by the pit wall structures."

7. In DCD Section 9.1.1.2.1, added the sentence "The new fuel storage rack array center-to-center spacing of nominally 10.9 inches provides a minimum separation between adjacent fuel assemblies sufficient with neutron absorbing material to maintain a subcritical array."

8. In DCD Section 9.1.1.2.1, deleted "racks are purchased equipment. The purchase specification for the new fuel storage racks will require the vendor to perform confirmatory dynamic and stress analyses."

9. In DCD Section 9.1.1.2.1, changed future tense "will be done by the combined operating license (COL) applicant" to present perfect tense "has been done by DC applicant".

10. In DCD Section 9.1.1.2.1, deleted "and is braced as required to the pit wall structures."

11. In DCD Section 9.1.1.2.1 and Section 9.1.1.3, changed the maximum uplift load from 907 kilograms (kg) (2000 pounds (lb)) to 1814 kg (4000 lb).

12. In DCD Section 9.1.1.2.1, added the weight of the fuel handling tool and the control rod assembly to the weight of the fuel assembly in the fuel drop analysis. This changed the total drop weight from 850 kg (1875 lb) to 919 kg (2027 lb).

13. In DCD Section 9.1.1.2.1, replaced "The crane and the attachment to the building structure …" with "The fuel handling machine …"

14. In DCD Section 9.1.1.3, changed future tense "will be done by combined operating license (COL) applicant" to present perfect tense "has been done by DC applicant."

15. In DCD Section 9.1.1.3, deleted “The new fuel storage rack is purchased equipment. The purchase specification for the new fuel storage rack requires a criticality analysis of the new fuel storage racks.”

16. In DCD Section 9.1.1.3, identified that venting of the neutron absorbing material is “considered in the detailed design of the storage rack.”

The staff confirmed that the changes to DCD Revision 17, Section 9.1.1 were consistent with the design changes identified in TR-44, Revisions 0 and 1. TR-44, Revision 3, was issued in May 2010, to update both the design and the design basis, and to address unresolved issues. TR-44, Revision 4, was issued in July 2010, to clarify the analyses conducted to address sliding of the racks. TR-44, Revision 5, was issued in August 2010, to incorporate the applicant’s new position on accidental fuel assembly drops over the new fuel pit.

9.1.1.2.1.2 Evaluation

TR-44, Section 2.8.5, Revision 0, indicated that there were two postulated fuel drop scenarios over the new fuel pit, both from a height of 91.4 cm (36 in) above the top of the new fuel storage rack. In request for additional information (RAI)-TR44-01, the staff requested that the applicant describe the fuel handling operations that lead to the assumed 91.4 cm (36 in) drop height. The staff also issued RAI-TR44-02 through RAI-TR44-07, requesting specific information related to the accidental drop analysis.

In August 2010, the applicant submitted TR-44, Revision 5, in which the applicant deleted all reference to postulated new fuel assembly drop scenarios over the new fuel pit. The basis for this deletion was submitted in a revised response to RAI-TR44-01, dated August 13, 2010. This is further supported by the applicant’s revised response to RAI-TR44-06, dated September 13, 2010. The applicant stated that the new fuel is moved from the rail car bay to a cell in the new fuel storage rack by the single-failure proof hoist. The same hoist then moves the fuel assembly from the cell in the new fuel storage rack to the new fuel elevator. This sequence is repeated for each new fuel assembly, 67 per outage, on an 18-month schedule. The applicant stated that the single-failure-proof hoist is designed to meet the requirements of NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” and is the only hoist capable of moving the new fuel above the operating floor. The applicant further stated that there are no safe shutdown systems or components currently housed in the new fuel pit or the resin transfer pump/valve room below it, and there are no criticality concerns for new fuel storage. The new fuel handling tool incorporates the same design features as the spent fuel handling tool (SFHT), which prevents inadvertent release of the new fuel assembly during handling operations. Based on the information in the revised RAI responses, the staff finds that the applicant has adequately justified that a new fuel assembly drop is unlikely and does not require analysis. RAI-TR44-02, RAI-TR44-03, RAI-TR44-04, RAI-TR44-05, and RAI-TR44-07 are no longer relevant, and are not discussed in this report.

As indicated in Table 2-3 of TR-44, one of the fuel handling accident loads that needs to be considered is uplift force on the rack caused by a postulated stuck fuel assembly. TR-44 Section 2.8.3 states: “An evaluation of a stuck fuel assembly, leading to an upward load of 2,000 lb has been performed. The results from the evaluation show that this is not a bounding condition as the local stresses do not exceed 2,500 psi.” The staff determined that the information provided was not sufficient for the staff to reach a conclusion that this load had been adequately considered. In RAI-TR44-08, the staff requested the applicant to provide a detailed description of the assumptions, the analyses conducted, the results obtained, and the basis for the conclusion that this is not a bounding condition.

In a letter dated June 7, 2007, the applicant stated that a nearly empty rack with one corner cell occupied is subject to an upward load of 907 kg (2000 lb), which is assumed to be caused by the fuel sticking while being removed. The ramification of the loading is two-fold:

1) The upward load creates a force and a moment at the base of the rack;
2) The loading induces a local tension in the cell wall.

The applicant attached a calculation documenting the maximum stress in the rack cell structure due to a postulated stuck fuel assembly. This local stress is well below the yield stress of the cell wall material (i.e., 207 MPa (30,000 psi)).

The basis for resolution of this RAI is similar to that of RAI-TR54-14, which is discussed in Section 9.1.2 of this report. The applicant re-defined the uplift force to be 1814 kg (4,000 lb), and showed that the induced stress is still well below the material yield stress. The applicant made appropriate changes in TR-44, Revision 1, and identified proposed changes to DCD Revision 17. The staff found this acceptable. However, the applicant’s proposed changes to DCD Revision 17 were not completely implemented. During the June 2-3, 2010, regulatory audit, the staff asked the applicant to submit a supplemental RAI response, addressing the omission. In a letter dated July 20, 2010, the applicant submitted a supplemental response, explaining that the paragraph omitted from the DCD revision discussed the new fuel handling crane, which has been superseded. Therefore, it is no longer relevant. The staff finds this explanation acceptable. Therefore RAI-TR44-08 is resolved.

The staff noted that insufficient descriptive information was included in TR-44, Revision 0, to permit an adequate review of the structural/seismic analysis of the new fuel rack. In RAI-TR44-09, the staff requested the applicant to provide descriptive information, including plans and sections showing the new fuel rack and vault walls. All of the major features of the rack, including the cell walls, baseplate, pedestals, bearing pads, neutron absorber sheathing, any impact bars, welds connecting these parts, and any other elements in the load path of the rack should be shown on one or several sketches.

In a letter dated July 17, 2007, the applicant stated that TR-44 Figures TR-44-9.1 through TR-44-9.5 provide additional descriptive information on the new fuel rack and new fuel storage pit floor and walls. The staff confirmed that appropriate additions were made in TR-44, Revision 1. In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-9, because several dimensions on drawings had been updated. The applicant stated that the most recent change to DCD Figure 9.1-1 (Sheet 1 of 2) is included in the response to RAI-TR44-17 Revision 3, which was also submitted in a letter dated July 20, 2010. The staff reviewed the change to DCD Figure 9.1-1 and found it acceptable. RAI-TR44-09 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text.

The staff noted that TR-44, Revision 0, did not provide sufficient data regarding the input loads used for the seismic analysis of the new fuel rack. The staff issued RAI-TR44-11, which reads as follows:

a. Floor response spectra (X, Y, and Z - vertical directions) at or near the elevation of the top of the fuel rack and near the bottom of the fuel rack or vault floor corresponding to the damping value used for the analysis.

b. An explanation of why the envelope of these two sets of spectra were not used.

c. An explanation of the range of soil and rock properties used in enveloping the seismic floor spectra. Given that the certified DCD is applicable to a hard rock site and the location of e are these range of soil/rock properties specified for confirmation by a future COL applicant?

d. For the synthetic time histories, plots of the three time histories, the cross correlation coefficients, the comparisons of the spectra from the synthetic time histories to the enveloped target response spectra, and the comparisons of the power spectral density plots to the target power spectral density function associated with the target response spectra.

e. Which time history was used (displacement, velocity, or acceleration)? Were all three directions input simultaneously? Was gravity included in the time history analysis?

In a letter dated May 3, 2007, the applicant provided the following response:

a) Floor response spectra (X, Y, and Z - vertical directions) near the elevation of the bottom of the new fuel storage vault corresponding to the damping value used for the analysis are provided in the attachment to this response. No floor response spectra are provided near or at the elevation of the top of the new fuel rack.

   The ASB99 floor response spectra (FRS) represent the enveloping response spectra for the auxiliary and shield building (ASB) at elevation 99 feet (ft) for a range of soil/rock condition. FRS of various soil/rock analyses were first enveloped for various locations of the ASB. All of the ASB locations at elevation 99 ft were then grouped and enveloped to develop the ASB99 floor response spectra.

b) It is probable that the floor response spectra will be revised for various reasons and that a revision to the new fuel storage rack structural/seismic analysis report (TR-44) will be required. The methodology for developing the spectra is described in TR-44-11 a, d and e responses.

c) The range of soil and rock conditions for which the seismic floor spectra applies is described in Westinghouse technical report (TR)-03, APP-GW-S2R-010, Revision 0, “Extension of NI Structures Seismic Analysis to Soil Sites.”

d) The synthetic time histories, the response spectrum curves, and the power spectral density plots for the Auxiliary and Shielding Building (ASB) at Elevation 99 feet are provided, with this response. The cross correlation coefficients for the three orthogonal components (East-West, North-South, and Vertical) of the ASB99 synthetic time histories are summarized in the table below.

Description	Cross Correlation Coefficient
East-West to North-South	-0.0414
East-West to Vertical	0.0088
North-South to Vertical	0.0536

e) Acceleration time histories are used as the input motion for the seismic analysis of the spent fuel racks. The acceleration input is defined by three orthogonal components, which are input and solved simultaneously. Gravity is also included in the time history analysis.

The staff found this RAI response acceptable because it adequately addressed all of the staff's questions. RAI-TR44-11 was initially resolved. However, subsequent to the initial resolution of this RAI, the applicant revised the seismic design loads twice. Therefore, during the June 2010 audit, the staff requested that the applicant update this RAI response to reflect the current seismic design loads for the new fuel rack. In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-TR44-11, updating the seismic design loads. The staff finds that the revised RAI response adequately describes the current seismic design loads for the new fuel rack. TR-44, Revision 3, includes the numerical results for the current design loads. Therefore, RAI-TR44-11 is resolved.

In RAI-TR44-12, the staff requested the applicant to address how the different impact stiffness values are determined for the fuel assembly-to-cell wall, rack-to-wall, and pedestal-to-bearing pad. In addition, since the impact forces can be greatly affected by the impact spring constant, the staff asked the applicant to address the sensitivity of the impact forces and rack responses to variations in these spring constants and whatever impact forces are imparted directly onto the cell walls or impact bars that are used?

In a letter dated July 5, 2007, the applicant stated that the impact stiffness values for the rack-to-wall and pedestal-to-bearing pad (concrete floor) are calculated as shown in Attachment 1 to the response. The fuel-to-cell wall impact stiffness is determined based on the solution for a simply supported circular plate under a concentrated load applied at its center, where the plate diameter is equal to the cell inner dimension and the plate thickness is equal to the cell wall thickness. The stiffness of the annular plate is then multiplied by the number of loaded storage cells for the new fuel storage rack, since the stored fuel assemblies are assumed to rattle in unison. A sensitivity study has not been performed specifically for the AP1000 new fuel rack to quantify the effect of variations in the impact stiffness values. However, sensitivity studies have been performed in the past for similar spent fuel rack applications submitted by HOLTEC, which employed the same method of computing the impact stiffness values, and the impact forces were found to be insensitive to small variations in the stiffness values provided that the integration time step was sufficiently small. There are no impact bars at the top of the new fuel storage rack. However, the new fuel storage rack is braced against the north and south walls of the new fuel storage pit by inserting stainless steel wedges in the interstitial space between the top of the new fuel storage rack and the pit opening.

The applicant subsequently changed the design of the new fuel rack to be free-standing in the new fuel pit. The steel wedges are no longer used. This is documented in TR-44, Revision 1. During the August 6-7, 2009 regulatory audit, the staff reviewed the updated calculations for the free-standing new fuel rack. The calculations support the applicant's conclusion in TR-44, Revision 1, that there are no credible impacts between the new fuel rack and the fuel pit walls, for the free-standing configuration.

Therefore, RAI-TR44-12 was resolved. A new issue arose related to new fuel rack sliding and potential wall impact, after the resolution of RAI-TR44-12. This is addressed in the discussion of RAI-SRP9.1.2-SEB1-02 in this report.

Section 2.2.2.2 of TR-44 describes the modeling of a single rack. It indicates that the rack cellular structure elasticity is modeled by a 3-D beam having three translational and three rotational degrees-of-freedom (DOFs) at each end so that two-plane bending, tension/compression, and twist of the rack are accommodated. In RAI-TR44-14, the staff requested the applicant to explain why shear stiffness/deformation is not also included and to provide more detailed information about how the beam model of the rack was developed, considering that it is an assembly of many square-celled structures welded at discrete locations.

In a letter dated April 13, 2007, the applicant stated that shear deformation is included in the rack dynamic model. The beam model of the rack was developed based on the applicable codes, standards and specifications given in Section IV(2) of the Nuclear Regulatory Commission (NRC) guidance on spent fuel pool (SFP) modifications entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, which states that "Design ... may be performed based upon the AISC specification or Subsection NF requirements of Section III of the ASME Code for Class 3 component supports." The rack modeling technique is consistent with the linear support beam-element type members covered by these codes.

The basis for resolution of this RAI is similar to that of RAI-TR54-23, which is discussed in Section 9.1.2 of this report. The staff confirmed that appropriate changes were made in TR-44, Revision 1. Therefore, RAI-TR44-14 was resolved.

Section 2.2.2.2 of TR-44 refers to Figure 2-2 for the dynamic beam model of a single rack. The text and figure do not adequately describe the model. The staff issued RAI-TR44-15, which reads as follows:

a. Define what each series of nodal degrees-of-freedom (DOFs) correspond to (i.e., nodes 1, 2; P1, P2, ...; q4, q5, ..., 1*, 2*, ...). While some of these may be deduced by judgment, the report should clearly define all of these.

b. Explain whether there are five (5) nodes and four (4) beams along the rack beam model to coincide with the five (5) nodes and four (4) elements of the fuel assemblies.

In a letter dated June 7, 2007, the applicant provided the following response:

a. The attached table defines the nodal DOFs for the dynamic beam model of a single rack as depicted in Figure 2-2 of the technical report.

b. The rack cell structure is modeled as a single beam between two nodes, which are located at the top of the rack and at the baseplate elevation. This is consistent with HOLTEC's standard model for seismic analysis of spent fuel racks, which has been reviewed and approved by the NRC on numerous dockets. Although there is not a one-to-one correspondence between beam nodes and fuel assembly nodes, fuel-to-cell

> wall impact loads, which can occur at elevation 0, 0.25H, 0.5H, 0.75H, and H (where H is the height of the cell structure), are properly transmitted to the rack beam in accordance with the methodology outlined in Reference 12 in COLA technical report APP-GW-GLR-026 Revision 0.

The basis for resolution of this RAI is similar to that of RAI-TR54-24, which is discussed in Section 9.1.2 of this report. The staff confirmed that appropriate changes were made in TR-44, Revision 1. Therefore, RAI-TR44-15 was resolved.

In RAI-TR44-16, the staff requested the applicant to explain whether only a full new fuel rack is considered in the simulation, or if several scenarios are considered; i.e., different fill ratios, from empty to full; and provide the technical justification if only a full rack is considered.

In a letter dated June 7, 2007, the applicant stated that the new fuel rack is assumed to be fully loaded with maximum weight fuel assemblies in all three simulations. This scenario bounds any partially loaded configuration since it: (1) maximizes the vertical compression and lateral friction loads on the support pedestals; and (2) produces the maximum rack displacements and fuel-to-cell wall impacts. The displacements are larger for a fully loaded rack, as opposed to a partially filled rack, because the dynamic model conservatively assumes that all stored fuel assemblies rattle in unison. Hence, the momentum transferred between the rattling fuel mass and the new fuel rack is the maximum for a fully loaded rack. For a partially filled rack, the decrease in rattling fuel mass outstrips the destabilizing effect of an eccentric fuel loading pattern.

The staff determined that a quantitative evaluation of this issue would most likely be required. During the August 2009 audit, the staff reviewed Draft Revision 3 to HOLTEC Calculation HI-2063492, which includes analysis of a partial loading case. The staff requested that the applicant revise its RAI response to reflect this additional loading case, and also to include these results in the next revision of TR-44.

The applicant formally submitted Revision 3 to TR-44 in May 2010, which includes the results of the partial loading case. At the June 2010 audit, the staff and applicant discussed these results, which indicate that partial loading generates the limiting conditions for certain loading cases. The staff noted that the applicant’s current RAI response indicates that the partial loading case is not controlling, and that the DCD does not identify that a partial loading was considered. In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-16, correcting the statement about the partial loading case, and also proposing a DCD change. The staff finds the response acceptable because it adequately addresses the inconsistency identified by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff issued RAI-TR44-17 which reads as follows:

> What are the gaps and tolerances for the gaps between the fuel assembly and cell wall, and between the rack and vault wall? What are the assumed initial locations of the various components (fuel assemblies and rack) and what is the technical basis for this assumption? Were any studies done for different initial conditions (considering tolerances); if not, explain why it was not necessary. Are there requirements in the DCD to ensure that the assumed gaps (considering tolerances) are maintained throughout the operating license period?

In a letter dated July 17, 2007, the applicant stated that all gaps between fuel assemblies and cell walls and between the rack and vault walls are set to match the nominal gaps provided on the layout drawing. The applicant attached a table summarizing the gap information used in the dynamic analyses. The applicant also stated that the new fuel storage rack is braced against the north and south walls of the new fuel storage pit by inserting stainless steel wedges in the interstitial space between the top of the new fuel storage rack and the new fuel storage pit opening. Fuel is assumed centrally located in the cell. This is conservative, since minimizing the gap on one or two walls will generally produce a larger hydrodynamic coupling effect. Some numerical studies were done on other rack projects; the results generally showed a small influence on results. A larger influence occurs if the gaps are assumed to be displacement dependent, rather than always being held constant at their initial value. The neglect of this effect is conservative.

The applicant further stated that, once the new fuel rack is installed, the "as-built" gaps are reconciled with the gaps initially used for analysis by evaluation of the numerical results and the predicted motions. The new fuel rack will be positioned in the new fuel storage pit in accordance with the gap information provided in the table attached to the July 17, 2007, letter. The only way the gaps would change over time would be by the action of a seismic event. COL applicants will have a procedure in place to address measurement of the post design-basis seismic event gaps, and to evaluate the acceptability of the configuration or to take appropriate corrective actions. The applicant proposed that the following statement be added to TR-44:

> Per DCD Subsection 3.7.5.2, Combined License applicants will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. An activity will be to address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit and to take appropriate corrective actions.

During the August 2009 audit, the staff discussed the need to revise the HOLTEC drawing depicting the design gaps and tolerances, to reflect the change in the new fuel rack design to be free-standing.

During the June 2010 audit, the staff confirmed the HOLTEC drawings have been updated; and also requested the applicant to clarify the dimension and gap information for the new fuel racks in three related RAIs (RAI-TR44-09, RAI-TR44-017, and RAI-TR44-25), to be consistent with the current design basis. The staff noted that DCD Figure 9.1-1 needed to be updated. In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-17, updating dimension and gap information. The applicant also proposed a change to DCD Figure 9.1-1. The staff finds the response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.3.4.3 of TR-44, fourth bullet, develops the faulted (Level D) allowable maximum weld stress for the weld material. In RAI-TR44-22, the staff asked the applicant whether an allowable maximum weld stress based on the base metal was also developed. The staff noted that normally welds are checked for both weld material and base metal, as was done for Levels A and B in TR-44 Section 2.3.4.1.

In a letter dated June 7, 2007, the applicant provided a response that is essentially the same as its response to RAI-TR54-33 on the same topic. The detailed review is discussed in

Section 9.1.2 of this report. Based on that discussion, the staff found this RAI response acceptable. Therefore, RAI-TR44-22 was resolved.

Section 2.3.5 of TR-44 discusses dimensionless stress factors. It states that “R1 is the ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression).” In RAI-TR44-23, the staff requested the applicant to explain why the pedestals only resist compression, since horizontal forces are also generated due to friction during a seismic event. These forces could be quite high and also would introduce shear and moments into the pedestal and rack structure.

In a letter dated July 17, 2007, the applicant stated that Section 2.3.5 of TR-44 defines seven stress factors (R1 through R7), which correspond to the American Society of Mechanical Engineers (ASME) Code Section III, Subsection NF stress limits for Class 3 components. R1 is defined as the ratio of direct tensile or compressive stress on a net section to its allowable value. Since the new fuel rack is freestanding, the net cross section of the support pedestals can only be subjected to direct compressive stress. The applicant further stated that horizontal forces are generated due to friction between the support pedestals and the floor and that these forces produce shear and bending stresses in the pedestals. The shear and bending stresses in the support pedestals, as well as the combined compression and bending stress, are measured by the other six stress factors (i.e., R2 through R7), which are defined in Section 2.3.5 of TR-44. The staff reviewed the RAI response and found it acceptable, because the applicant clarified that shear force and moment due to friction have been calculated and evaluated against applicable code limits. Therefore, RAI-TR44-23 was resolved.

Some of the information provided in Section 2.8.2 (Rack Structural Evaluation) and Tables 2-6 through 2-14 (stress results) of TR-44 is not clear. The staff issued RAI-TR44-24, which reads as follows:

a. Section 2.8.2.1, 2nd paragraph, indicates that the tables also report the stress factors for the AP1000 new fuel storage rack cellular cross section just above and below the baseplate. This implies that the fuel cells continue below the baseplate. Explain.

b. The same paragraph refers to “pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number).” Explain what this means since the tables do not reflect this terminology.

c. The same paragraph refers to “ensures that the overall structural criteria set forth in Subsection 2.2.3 are met.” Structural criteria are not presented in Subsection 2.2.3.

d. Section 2.8.2.2 a. refers to a stress factor of 2.1516, which it states is given in the tables. However, no such stress factor is given, please explain. Also, are all cells welded to the baseplate on all four sides?

e. Section 2.8.2.2 b. indicates that a separate finite element model is used to check the baseplate to pedestal welds. Provide a short description of the model, computer code, loading, and location of the maximum tabulated stress in the weld referred to in Table 2-12.

f. Section 2.8.2.2 c. indicates that for calculation of cell welds, the fuel assemblies in adjacent cells are conservatively calculated by assuming that the fuel assemblies in adjacent cells are moving out of phase with one another. It then states that cell to cell weld calculations are based on the maximum stress factor from all runs. However, elsewhere in the report, it was stated that all of the fuel assemblies in the simulation are assumed to vibrate in phase. Provide more information to explain this.

g. Section 2.8.2.3 refers to Tables 2-6 through 2-13 for limiting thread stresses under faulted conditions for every pedestal. These tables do not seem to apply to pedestal thread shear stress. Therefore, clarify or correct this information.

h. For Table 2-6, Results Summary, please identify what rack component/element applies to each of the column headings (i.e., Max Stress Factor, Max. Shear Load, Max Fuel to Cell Wall Impact). Similarly, for Tables 2-11, 2-13, and 2-14, identify what rack component/element the table applies to.

i. Why is Table 2-14 labeled "Allowable Shear Stress for Level D"? This is inconsistent with the other tables. Explain.

In a letter dated July 17, 2007, the applicant provided the following response:

(a) The fuel cells do not continue below the baseplate. Stress factors are computed just above the baseplate, where the fuel cells are welded to the baseplate, and just below the baseplate where the support pedestals are welded.
Section 2.8.2.1 (2^{nd} paragraph, 2^{nd} sentence) will be revised as follows:

> "The tables also report the stress factors for the AP1000 new fuel storage rack cellular cross section just above the baseplate."

(b) The computer code DYNAPOST, which is listed in Table 2-15, computes the stress factors for the four support pedestals and for the cellular structure just above the baseplate based on the time history analysis results. For convenience, these five locations are identified as pedestal numbers 1 through 5 in the DYNAPOST output tables, which are not included in TR-44. Therefore, the sentence, "The locations above the base plate … are referred to as pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)" is not relevant to the report and will be deleted.

(c) The reference to Subsection 2.2.3 is a typographical error. The correct reference is Subsection 2.3.3.

(d) The factor of 2.1516 is not provided in the tables as stated in text.
Section 2.8.2.2 a. (2^{nd} paragraph) will be revised as follows:

> "Weld stresses are determined through the use of a simple conversion (ratio) factor (based on area ratios) applied to the corresponding stress factor in the adjacent rack material. This conversion factor is developed from the differences in base material thickness and length versus weld throat dimension and length." All fuel cells are welded to the baseplate on all four sides.

(e) The finite element code ANSYS is used to resolve the tension and compression stresses in the pedestal weld due to the combined effects of a vertical compressive load in the pedestal and a bending moment caused by pedestal friction. The compression interface between the baseplate and the pedestal is modeled using contact elements. The perimeter nodes on the pedestal are connected to the baseplate by spring elements in order to simulate tension in the weld. The maximum instantaneous friction force on a single pedestal from the rack seismic analysis is conservatively applied to the finite element model in the horizontal x- and y-directions simultaneously, along with the concurrent vertical load, at the appropriate offset location. The perimeter nodes on the pedestal are restrained to move only in the vertical direction so that the spring elements only resist bending. The limiting ANSYS results are combined with the maximum horizontal shear loads to obtain the maximum weld stress. The maximum weld stress reported in Table 2-12 occurs at the corner of the pedestal where the tensile stress in the weld due to bending is maximum.

(f) All stored fuel assemblies within a rack are assumed to rattle in phase for the seismic analysis of the new fuel rack using the HOLTEC proprietary computer code MR216 (a.k.a. DYNARACK). This analysis yields the maximum impact force between a single fuel assembly and the surrounding cell walls. When evaluating the weld connection between adjacent storage cells, the maximum fuel-to-cell impact force from the dynamic analysis is conservatively multiplied by a factor of 2 to consider out-of-phase fuel rattling.

(g) The reference to “Tables 2-6 through 2-13” in Section 2.8.2.3 is incorrect. The first sentence in Section 2.8.2.3 should be revised as follows: “Table 2-14 provides the limiting thread stress under faulted conditions.”

(h) In Table 2-6, the “Max. Stress Factor” column applies to the rack cell structure. The “Max. Vertical Load” and “Max. Shear Load” columns apply to a single rack pedestal. The “Max. Fuel-to-Cell Wall Impact” column provides the maximum impact force between a single fuel assembly and the surrounding cell wall at any of the five rattling fuel mass elevations (refer to Figure 2-5 of the report). Table 2-11 applies to the base metal adjacent to the baseplate to cell welds. Table 2-13 provides the shear stress in the cell to cell welds as well as the adjacent base metal. Table 2-14 applies to the pedestal internal threads.

(i) Table 2-14 should be labeled “Pedestal Thread Shear Stress” instead of “Allowable Shear Stress for Level D.” The allowable stresses reported in Tables 2-10 through 2-14 are Level D stress limits since the design basis ASB99 earthquake is a faulted condition (Level D).

The basis for resolution of this RAI is similar to that of RAI-TR54-36, which is discussed in Section 9.1.2 of this report. Based on that discussion and confirmation that the proposed changes were made in TR-44, Revision 1, the staff found this RAI response acceptable. Therefore, RAI-TR44-24 was resolved.

In the markup of the DCD, provided in Section 5 of TR-44, Revision 0, DCD Figure 9.1-1, new fuel storage rack, is identified for deletion. In RAI-TR44-25, the staff requested the applicant to explain why this figure was marked for deletion.

In a letter dated April 13, 2007, the applicant stated, “We are in agreement Revision 16 of the DCD will have a revised Figure 9.1-1 New Fuel Rack Layout. This figure will show the new fuel rack configuration in plan and elevation views identifying significant features and dimensions”.

The staff reviewed the RAI response and concluded that review of DCD Revision 16 would be necessary to determine if sufficient information is included in the revised Figure 9.1-1.

In TR-44, Revision 1, the applicant changed the new fuel rack design to be free-standing, which affects DCD Figure 9.1-1. This was discussed with the applicant and HOLTEC at the August 2009 audit. The applicant agreed to revise the figure as necessary to reflect the design change.

In a letter dated July 20, 2010, the applicant submitted a revised response to RAI-TR44-25, referencing its revised response to RAI-TR44-17, also submitted by letter dated July 20, 2010, for the updated dimension and gap information. RAI-TR44-25 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff noted that computer code MR216 (a.k.a. DYNARACK), as well as the other computer analysis codes used for the rack analyses, should have complete validation documentation, available for review during an audit. To streamline such an audit, in RAI-TR44-26, the staff inquired whether any of the computer codes have been previously reviewed and approved by the staff in other licensing reviews.

In a letter dated June 7, 2007, the applicant stated that computer analysis codes used to perform the seismic analysis of the new and spent fuel racks have been validated in accordance with HOLTEC's 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," quality assurance program. The validation documentation will be available for review during the audit. The validation documentation for the computer code MR216 has been previously submitted by HOLTEC International to the NRC staff for review and approval several times. Most recently it was reviewed by the NRC in 1998 in Docket 50-382 for the Waterford 3 Steam Electric Station.

The basis for resolution of this RAI is similar to that of RAI-TR54-39, which is discussed in Section 9.1.2 of this report. Based on that discussion, the staff found this RAI response acceptable. Therefore, RAI-TR44-26 was resolved.

In its review of TR-44, the staff did not identify any information related to inservice inspection of the new fuel rack. In RAI-TR44-27, the staff requested the applicant explain what provisions are provided for performance of inservice inspection of the rack, in accordance with 10 CFR 50.55a, "Codes and standards" and/or 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," as applicable.

In a letter dated June 7, 2007, the applicant stated that the new fuel rack is passive in nature. There are no moving parts on the new fuel rack, and it does not require any instrumentation. Therefore, there is no compelling need to perform inservice examination of the new fuel rack. Nonetheless, the new fuel rack can be accessed from above by way of an empty storage cell location(s) to enable the performance of inservice examination, as mandated by 10 CFR 50.55a(g)(3) for ASME Code Class 3 component supports. At the base of each storage cell (except at the four designated lifting locations), there is a 15.24 cm (6 in) diameter thru-hole in the baseplate, which provides access below the baseplate. The new fuel rack contains new fuel only during a short period prior to refueling. When it does not have new fuel, it could be lifted from the new fuel storage pit for inspection.

In summary, the new fuel rack is designed to provide access to surfaces that may come in contact with new fuel assemblies and to the support pedestals beneath the baseplate to support inservice examinations as needed.

From the information provided by the applicant, the staff concluded that there is ample access to the new fuel rack for inservice inspection. Therefore, RAI-TR44-27 was resolved.

The treatment of the new fuel storage rack as a safety class/seismic Category I component appears to represent a departure from past practice in the nuclear power industry. The draft update to RG 1.29, “Seismic Design Classification,” (DG-1156) does not identify new fuel storage racks as seismic Category I. In RAI-TR44-28, the staff requested the applicant to: (1) describe the technical basis for treating the new fuel storage rack as a safety class/seismic Category I component; and (2) explain how the safety significance of the AP1000 new fuel storage rack differs from prior nuclear power plant designs.

In a letter dated June 7, 2007, the applicant provided the following response:

1) We understand that both Regulatory Guide 1.29, Revision 3 and draft update to RG 1.29 (DG-1 156) do not identify new fuel storage racks as seismic Category I. However, the Westinghouse decided that all racks in the AP1000 plant would be seismic Category I. Holtec has designed and fabricated new fuel storage racks to seismic Category I. There is no additional analysis or fabrication cost to have the new fuel storage rack as seismic Category I.

2) The safety significance of the AP1000 new fuel storage rack does not differ from prior nuclear power plant designs. It is both the applicant's and HOLTEC's position that the form, fit and function of the AP1000 new fuel storage rack is the same as new fuel racks in operating PWRs.

On the basis that the applicant has invoked more stringent seismic design requirements than RG 1.29, the staff found that its technical approach is acceptable. Therefore, RAI-TR44-28 was resolved.

Section I, “Introduction,” was revised in TR-44, Revision 1 to add: “Per DCD Subsection 3.7.5.2, COL applicants will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. An activity will be to address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit and to take appropriate corrective actions.”

The staff noted that DCD Section 3.7.5.2 does not discuss the need for COL applicants to prepare site-specific procedures for checking the gaps between the new fuel rack and walls of the new fuel storage pit following an earthquake. In RAI-SRP9.1.2-SEB1-01, the staff requested the applicant to explain how this requirement is conveyed to the COL applicants.

In its response dated February 24, 2009, the applicant proposed a revision to Section 3.7.5.2 of the DCD, requiring COL applications to include in their post-earthquake procedure a requirement to check the gaps between the new fuel rack and the new fuel pit walls and to take appropriate actions to restore the design gaps. Following the staff’s August 2009 audit, the applicant submitted a revised response on August 31, 2009. The staff found this revised RAI response acceptable, because it more clearly specifies the actions to be taken following an earthquake. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In its review of TR-44, Revision 1, the staff noted that Section 2.8.1.4 “Impact Loads” was not revised, even though shims between the new fuel rack and the fuel pit wall have been deleted from the design. Quoting from TR-44, Revision 1, Section 2.8.1.4, “The maximum impact load from the set of shims that close the north-south gaps at the top of the rack is summarized in Table 2-8.” The staff also noted that the maximum rack-to-wall impact load in Table 2-8 increased from 50,802 kg (112,000 lb) in TR-44, Revision 0, to 69,853 kg (154,000 lb) in TR-44, Revision 1.

In RAI-SRP9.1.2-SEB1-02, the staff requested the applicant explain why the impact load increased, and describe how the design of the new fuel rack and the new fuel pit wall were evaluated for the significant increase (35 percent) in the impact load, in addition to other concurrent loadings.

In a letter dated April 1, 2009, the applicant submitted its response to RAI-SRP9.1.2-SEB1-02. The applicant stated that all references to shims will be deleted from TR-44. The applicant described the basis for the increased impact force, and clarified that the impact is the base plate against the wall, not the top of the rack against the wall. The applicant also stated that this impact only occurs for an unrealistically low coefficient of friction (COF) (0.2) between two dry steel surfaces. At realistic values (>0.5), the new fuel rack does not slide, and there is no rack-to-wall impact. The staff concluded that the detailed calculation(s) leading to the conclusion that there are no credible impacts between the new fuel rack and the fuel pit walls needed to be audited. At the August 2009 audit, the staff reviewed the detailed calculation, and found it acceptable to support the conclusion of no impact. The applicant agreed to revise TR-44 and Calculation APP-FS01-S3C-001 (HOLTEC Calculation HI-2063492), to clarify that there are no credible impacts between the new fuel rack and the new fuel pit walls, and to define the technical basis for this conclusion.

In May 2010, the applicant informed the staff that the COF could be as low as 0.24 because the actual surface condition is now steel on concrete, not steel on steel as it was previously. The applicant performed new calculations and evaluation of sliding and impact using a COF of 0.24. At the June 2010 audit, the applicant presented the results of the analysis for a 0.24 coefficient to the staff. TR-44, Revision 3, submitted just prior to the audit, includes the results of the new calculations. The staff reviewed the presentation material and related section of TR-44, Revision 3. The staff pointed out several apparent inconsistencies in the results presented, possibly indicative of an incorrect COF. Upon examination, HOLTEC confirmed that an incorrect COF had been used in one of the cases analyzed. During the audit, HOLTEC corrected the input file, re-executed the analysis, and presented the corrected results. The staff reviewed the corrected results, and found them acceptable.

The new results, for a 0.24 COF, show considerable sliding of the new fuel rack, but impact is avoided by increasing the minimum gap between the new fuel rack and the new fuel pit walls. The applicant agreed to revise the response to RAI-SRP9.1.2-SEB1-02 to describe the updated evaluation of sliding and impact, and also to identify necessary changes to TR-44.

In a letter dated July 30, 2010, the applicant submitted a revised response to RAI-SRP9.1.2-SEB1-02. The applicant stated that it had re-evaluated the range of appropriate friction values, to ensure that the interface between the new fuel storage rack and the new fuel pit floor is accurately and conservatively represented. The applicant concluded that the appropriate credible lower-bound COF is presented in Run Number 5 (COF=0.24). Additionally, Run Number 1 (COF=0.2) has been demonstrated to be non-credible, as noted above. It will be maintained in the supporting calculations, but for clarity it will be eliminated from the tables in the next revision of TR-44.

The applicant also attached TR-44, Revision 4 (July 2010) to the RAI response. Revision 4 clarifies what run numbers apply to the design basis and provides the detailed results for the credible lower-bound COF (COF=0.24). The report eliminates reference to the non-credible case for Run Number 1 (COF=0.2).

The staff reviewed TR-44, Revision 4, and confirmed that it includes appropriate new information related to Run Number 5 and that Run Number 1 has been deleted. The staff finds this acceptable to properly document the design basis for the new fuel rack. Therefore, RAI-SRP9.1.2-SEB1-02 is resolved.

Section 2.8.4.1, “Cell Wall Buckling Evaluation,” was revised in TR-44 Revision 1. The buckling equation and assumed rectangular plate boundary conditions were changed. The rectangular flat plate model representing the lower cell wall region was changed to clamped on all four edges. In the Revision 0 calculation, simple support was assumed. The staff determined that only one edge can truly be treated as clamped, and the other three edges can rotate somewhat due to the flexibility of the adjacent sections.

The staff issued RAI-SRP9.1.2-SEB1-03, which reads as follows:

> Section 2.8.4.1, "Cell Wall Buckling Evaluation" was revised in TR-44 Revision 1. A different buckling equation and different boundary conditions are indicated. The rectangular flat plate model representing the lower cell wall region is now assumed to be clamped on all 4 edges, Considering that only 1 edge can truly be treated as clamped, and the other 3 edges can rotate somewhat due to the flexibility of the adjacent sections, the staff requests Westinghouse to provide the technical basis for changing the boundary conditions to clamped on all 4 edges. Also, identify the minimum acceptable factor of safety and the technical basis for its selection.
>
> The staff notes that for K = 7.23, the revised σ_{cr} should be 15,600 psi, not 13,100 psi. Also, there is a typographical error: "31,100" should have been "13,100." The staff requests Westinghouse to correct the text of Section 2.8.4.1 accordingly.
>
> The staff also requests Westinghouse to identify the factor of safety based on the Rev 0 estimate of σ_{cr} = 7,090 psi.

In a letter dated April 14, 2009, the applicant submitted its response to RAI-SRP9.1.2-SEB1-03, in which it provided its technical basis for the revised calculation of buckling for the cell wall. The staff reviewed the response and determined that the information provided was insufficient, and that a significantly expanded technical basis would be needed before the staff could accept the cell wall buckling calculation. At the August 2009 audit, HOLTEC informed the staff that it was conducting a detailed nonlinear analysis of the bottom of the rack for vertical compressive load, and presented the ANSYS computer model and preliminary results. The staff found this to be a considerable analytical improvement. During the June 2010 audit, the staff reviewed

HOLTEC's final results of the ANSYS buckling evaluation of the cell walls, at the base of the new fuel rack. The calculation shows that a 1.5 factor of safety, in accordance with the acceptance criterion in ASME Code Section III, Subsection NF, has been achieved. The staff finds the analytical method used and the results obtained to be acceptable, based on its detailed review of HOLTEC's calculation. Therefore, RAI-SRP9.1.2-SEB1-03 is resolved.

9.1.1.2.1.3 Conclusion

The staff has conducted a detailed review of TR-44, which addresses DCD Revision 15 COL Information Item 9.1-1: "Perform a confirmatory structural dynamic and stress analysis for the new fuel rack, as described in AP1000 DCD Subsection 9.1.1.2.1. This includes the structural adequacy of the proposed AP1000 new fuel storage rack under postulated loading conditions and effects on the structure described in Subsection 3.8.4." The staff finds the new fuel rack design, as described in TR-44, Revision 5, to be acceptable. On the basis of its review, the staff concludes that the substance of the COL information item is completely addressed by TR-44, and that this COL information item is no longer needed.

In its previous evaluations of AP1000 DCD Section 9.1.1, the staff identified acceptance criteria based on the design's meeting relevant requirements in 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 2, "Design Bases for Protection Against Natural Phenomena," and in GDC 4, "Environmental and Dynamic Effects Design Bases." The above evaluation concludes that the new fuel rack design meets these 10 CFR Part 50 requirements.

9.1.1.2.2 Fuel Handling Crane Change

The applicant proposed the following changes for the fuel handling crane:

1. Deleted references to the fuel handling jib crane and replaced them with references to the new-fuel handling crane in DCD Sections 9.1.1.1; 9.1.1.2; 9.1.1.2.1; and 9.1.1.3.

2. The capacity of the fuel handling crane was limited to 907 kg (2000 lb), the applicant is now proposing that the capacity of the fuel handling crane be limited to lifting a fuel assembly, control rod assembly, and handling tool.

3. The uplift capability of the new-fuel handling crane was increased from 907 kg (2000 lb) to 919 kg (2027 lb) in DCD Section 9.1.1.3.

However, in response to RAI-SRP9.1.4-SBPB-01, the applicant stated in a letter dated June 26, 2008, that the function of moving new fuel will be transferred to the fuel handling machine (FHM) and that the new-fuel handling crane will be eliminated.

The evaluation of this change is reviewed in Section 9.1.4 of this report. The staff determined that the changes made to DCD Section 9.1.1 are conforming changes that do not impact the staff's safety evaluation of DCD Section 9.1.1. Therefore, the staff finds the proposed change acceptable.

9.1.1.2.3 New Fuel Criticality Analysis

9.1.1.2.3.1 Summary of Technical Information

In the certified DCD Revision 15, Section 9.1.1, "New Fuel Storage," it is stated in Section 9.1.6.2 that the COL applicant is responsible for a confirmatory criticality analysis for the new fuel rack, as described in Section 9.1.1.3. This is COL Information Item 9.1-2. In DCD Revision 17, the applicant proposed to perform the confirmatory criticality analysis so that COL action is no longer necessary. DCD Section 9.1.1.3 is revised to reflect that the criticality analysis is now complete and Section 9.1.6.2 is revised to state that the COL information requested in this section has been completely addressed in TR-67, and the applicable changes are incorporated into the DCD. The applicant stated that no additional work is required by the COL applicant. The technical details of the criticality analysis for the AP1000 new fuel storage design is presented in TR-67. This report provides the technical support for the changes found in Section 9.1.1 of the DCD. The staff's review of the criticality analysis of AP1000 new fuel storage includes DCD Revision 17 Section 9.1.1 and TR-67.

The staff based its review of the AP1000 new fuel storage on the information in the DCD and the TRs referenced by the applicant. The review was limited in scope to the changes to the new fuel storage criticality analysis of DCD Revision 15 (NUREG-1793) as presented in Revision 17. The staff conducted its review of the criticality analysis of the new fuel storage in accordance with the guidelines provided in Section 9.1.1 of NUREG-0800.

10 CFR Part 50, Appendix A, provides a list of the minimum design requirements for nuclear power plants. According to GDC 62, "Prevention of Criticality in Fuel Storage and Handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes.

The regulations regarding criticality for the AP1000 new fuel storage rack are as follows:

- 10 CFR 50.68(B)(2), "Criticality accident requirements," states the maximum K_{eff} value, including all biases and uncertainties, must not exceed 0.95, at a 95 percent probability and a 95 percent confidence level, with full density unborated water.

- 10 CFR 50.68(B)(3) states the maximum K_{eff} value, including all biases and uncertainties, must not exceed 0.98, at a 95 percent probability and a 95 percent confidence level, with optimum moderation and full reflection conditions.

- 10 CFR 50.68(B)(7) states the maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent (w%) U-235.

9.1.1.2.3.2 Evaluation

The criticality analysis of new fuel storage racks for the AP1000 design is presented in TR-67.

Methodology

The analysis methodology uses SCALE-PC, a personal computer version of the SCALE-4.4a code system with the updated SCALE-4.4a version of the 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross-section library.

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSA25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on personal computers.

Validation of SCALE-PC for fuel storage rack analyses is based on the analysis of representative critical experiments from two experimental programs: the Babcock & Wilcox (B&W) experiments in support of Close Proximity Storage of Power Reactor Fuel and the Pacific Northwest Laboratory Program in support of the design of Fuel Shipping and Storage Configurations. DCD Revision 15 and "Criticality Safety Criteria" (TANS Volume 35, page 278 dated 1980) were useful in updating pertinent experimental data for the Pacific Northwest Laboratory experiments. The validation of SCALE-PC was limited to the 44 group library provided with the SCALE-PC version 4.4a package (Westinghouse calculation CN-CRIT-206).

The approach used for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of "Criticality Safety Criteria (TANS Volume 35, page 278 dated 1980) and DCD Revision 17. For a given KENO-calculated value of K_{eff} and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed. By this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

Assumptions

The assumptions used for the AP1000 new fuel storage rack are as follows:

- The Westinghouse AP1000 17x17 fuel assembly was modeled as the design basis fuel assembly with an enrichment equal to 5.0 w% U-235.

- Storage cell material and Metamic® poison material were modeled with a length of 426 cm (168 in). The storage cell material and Metamic® poison above and below the active fuel length were conservatively omitted.

- Fresh fuel assemblies were conservatively modeled with a UO_2 density of 10.686 gram per cubic centimeters (g/cm^3) (97.5 percent of theoretical density). This translates into a pellet density equal to 98.6 percent of theoretical density with a 1.1 percent dishing (void) fraction.

- All fresh fuel assemblies were conservatively modeled as containing solid right cylindrical pellets and uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs that incorporate lower enrichment blanket or annular pellets.

- All fuel assemblies were conservatively modeled as containing no burnable absorber material.

Design Input

The fuel assembly modeled in this analysis is the Westinghouse AP1000 17xl7 fuel assembly present in Figure 2-1 of TR-67. The bottom elevation of the Metamic® poison panel (436.9 cm (172 in) long) conservatively covers the active fuel length (427 cm (168 in)) with a 10.16 cm (4 in) overlap (5.08 cm (2 in) overlap on each end of the active fuel). Design data related to the AP1000 new fuel storage rack were obtained from HOLTEC. In addition, HOLTEC supplied Westinghouse a pressurized-water reactor (PWR) rack data sheet, which provides detailed new fuel storage rack design information (HOLTEC International Letter Number 1540001.doc, "PWR Rack Data Sheet," Revision 4, April 27, 2006).

Westinghouse general arrangement and concrete outline drawings were used to determine the new fuel storage vault geometry. A plan view of the AP1000 new fuel storage rack is shown in Figure 2-2 of TR-67. The AP1000 new fuel storage rack is located inside a concrete room (vault) in the AP1000 Auxiliary Building. The AP1000 new fuel storage rack is centered inside the vault and is an 8x9 array of storage cells, which provides 72 total storage locations. A hatch lid is provided for the vault for security, and for foreign material exclusion (FME).

The individual storage cells of the AP1000 new fuel storage rack are centered on a nominal pitch of 27.7 cm (10.9 in). Each storage cell consists of an inner stainless steel box, which has a nominal inside dimension of 22.4 cm (8.8 in) and is 0.19 cm (0.075 in) thick. Metamic® panels are attached to the outside surfaces of all storage cells except for the outside cell walls directly facing the North and South walls of the vault. No poison is required on these outside cell faces since there is just a small amount of space between the rack and storage vault concrete. However, poison is required on the outside cell faces in the East and West directions (see Figure 2-2 in TR-67) to mitigate the effects of an inadvertent placement of a fuel assembly outside of the rack, but within the vault on these two sides. Each Metamic® poison panel is held in place and is centered on the surface of the stainless steel box by an outer stainless steel sheathing panel. There is a small void space between the sheathing and the Metamic® panel. The dimensions of the Metamic® poison panel are 19.06 cm (7.5 in) in width by 0.27 cm (0.106 in) in thickness. The sheathing panels are 0.089 cm (0.035 in) in thickness.

Each storage cell is nominally 506.7 cm (199.5 in) long, and it rests on top of a base plate whose top is 12.7 cm (5 in) above the concrete floor. As stated above, each Metamic® poison panel is 436.9 cm (172 in) long overlapping the 426.7 cm (168 in) active fuel length. The Metamic® poison material is nominally a mixture of 31 w% B_4C and 69 w% aluminum.

KENO Model and Assumptions

The KENO V.a model is a three-dimensional representation of the AP1000 new fuel storage rack and vault. The 17x17 fuel assemblies are explicitly modeled as shown in Figure 2-1 of TR-67, and each assembly is fully enriched with 5 w% U-235 over the entire length of the active fuel (426.7 cm (168 in).

The 8x9 array of storage cells is modeled with the active fuel of a 17x17 fuel assembly in each location. The 8x9 array of storage cells is centered with the four walls of the concrete vault. The top of the vault is conservatively modeled without a lid. This is a conservative omission because a metal lid would absorb neutrons. Also included in the criticality model is the reflection provided by a postulated 30.48 cm (12 in) of full density water.

The fuel rod, guide tube, and instrumentation tube claddings are modeled with Zircaloy. This is conservative with respect to the Westinghouse ZIRLO product, which is a Zirconium alloy containing additional elements including Niobium. Niobium has a small absorption cross section, which causes more neutron capture in the cladding regions resulting in a lower reactivity. Therefore, this criticality analysis is conservative with respect to fuel assemblies containing ZIRLO cladding in fuel rods, guide tubes, and the instrumentation tube.

There are no burnable absorbers modeled in any of the fuel assemblies. The Zirconium grid straps are conservatively omitted. The 8x9 array of storage cells and vault are modeled at room temperature conditions (20 °C), and the system reactivity is evaluated at 11 moderator densities ranging from 1.0 g/ cm^3 down to 0.001 g/cm^3. A total of 1.2 million neutron histories are modeled in 1003 generations with 1200 neutrons per generation. It is noted that all KENO V.a results for the first three neutron generations are skipped to eliminate preliminary estimates of the system reactivity.

The methodology bias and uncertainty are discussed and evaluated in “Criticality Safety Criteria,” TANS Volume 35, page 278, dated 1980. The results of these KENO calculations showing final K_{eff} values (including bias and uncertainties) versus water density are given in Table 2-2 of TR-67. During a Regulatory Audit on November 16, 2006, the staff reviewed the applicant’s calculation, “AP1000 New Fuel Storage Rack Criticality Analysis,” which includes the uncertainty calculations as well as the final summation of uncertainties to be applied when calculating K_{eff} values.

Hypothetical Fuel Assembly Drop and Impact on Criticality Analysis

It is possible to drop a fresh fuel assembly into or on top of a storage cell in the AP1000 new fuel storage rack as described in Section 9.1.1.2.1 C of DCD Revision 17. In the event that the dropped fuel assembly hits the top of a storage cell, the applicant’s analyses indicate that neither the Metamic® nor the active fuel is adversely impacted. The applicant states there is no degradation of the criticality safety margin as a result of dropping a fuel assembly on top of a storage cell. The staff reviewed the applicant’s analyses and agrees with the approach and conclusions.

To conservatively bound the resulting deformation on the base plate following a drop of fuel assembly straight through an empty cell impacting the rack baseplate, the bottom elevations of 25 fuel assemblies were lowered by 12.7 cm (5 in). Even with the bottom elevation of the active fuel in 25 fuel assemblies lowered by 12.7 cm (5 in), the criticality design criteria given in Section 2.0 are still met. This conclusion is based upon the observation that the AP1000 new fuel storage rack is normally dry and not flooded with water. The applicant notes that for this hypothetical dropped fuel assembly accident the AP1000 new fuel storage rack would need to contain at least several feet of water in the bottom of the AP1000 new fuel storage rack before the criticality design basis limits would be exceeded. The accident scenario analyzed is beyond the design basis. Based on its review of the analysis, the staff finds the applicant’s design meets the criticality design basis limits for a hypothetical dropped fuel assembly.

Evaluation Results

Figure 2-2 of TR-67 displays the final K_{eff} values (including all biases and uncertainties) versus water density for the AP1000 new fuel storage rack. The maximum fresh fuel enrichment limit for the AP1000 new fuel storage racks is determined to be 5.0 w% U-235 since the final K_{eff} values at this enrichment are less than 0.98 at optimum moderation conditions and less than 0.95 at fully flooded conditions, assuming no soluble boron.

The staff has reviewed the changes submitted in TR-67, Revision 0. Given that: (1) the maximum K_{eff} value, including all biases and uncertainties, is less than 0.95 with full density unborated water; (2) the maximum K_{eff} value, including all biases and uncertainties, is less than 0.98 with optimum moderation and full reflection conditions; and (3) the maximum enrichment of fresh fuel assemblies is less than or equal to 5.0 w/o U-235, the AP1000 new fuel storage rack fully loaded with Westinghouse AP1000 17x17 fuel assemblies with an enrichment less than or equal to 5.0 w/o U-235 satisfies the criticality safety criteria specified in 10 CFR Part 50.68 and GDC 62 and, therefore, is acceptable.

9.1.1.2.3.3 Conclusion

The staff has reviewed the DCD Section 9.1.1 changes provided by DCD Revision 17 and supported by TR-67 submitted by the applicant, describing the new fuel storage racks, the criticality analyses performed, and the methods used. Based on this review, the staff concludes that the appropriate documentation was submitted and that the criticality aspects of the new fuel storage racks meet the requirements of GDC 62 related to the prevention of criticality.

9.1.1.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the applicant's application for design certification (DC) met the requirements of Subpart B to 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants" that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of AP1000 DCD, Section 9.1.1, the staff identified acceptance criteria based on the design meeting relevant requirements in GDC 2, GDC 4, GDC 5, "Sharing of Structures, Systems and Components," GDC 61, "Fuel Storage and Handling and Radioactivity Control," in GDC 62, and in GDC 63, "Monitoring Fuel and Waste Storage." The staff found that the AP1000 new fuel storage design was in compliance with these requirements, as referenced in NUREG-0800 Section 9.1.2 and determined that the design of the AP1000 new fuel storage, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 new fuel storage as documented in AP1000 DCD against the relevant acceptance criteria as listed above and in SRP, Sections 9.1.1, and 9.1.2. The staff finds that the applicant's proposed changes do not affect the ability of the new fuel storage to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD. The staff concludes that the AP1000 new fuel storage design continues to meet all applicable acceptance criteria. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes incorporated into Revision 17 contribute to the increased standardization of the certification information in the AP1000 DCD and, thus, meet the requirements of 10 CFR 52.63(a)(1)(vii). Therefore, the staff finds that the proposed changes to AP1000 Section 9.1.1 are acceptable.

9.1.2 Spent Fuel Storage

9.1.2.1 Summary of Technical Information

Section 9.1.2, "Spent Fuel Storage" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revision 17 the applicant has proposed to make the following changes to Section 9.1.2 of the certified design:

1. Spent fuel rack design change. The basis for this change is documented in TR-54, "Spent Fuel Storage Racks Structure and Seismic Analysis," APP-GW-GLR-033, Revision 4 of June 2010. The applicant added detailed design information for the spent fuel racks and proposed several changes throughout DCD Section 9.1.2 to reflect the changes addressed in TR-54.

2. SFP water level increase. The basis for this change is documented in TR-121, "Spent Fuel Pool Water Level and Dose," APP-GW-GLN-121, of May 2007. The applicant proposed several changes throughout DCD Section 9.1.2 to reflect the changes addressed in TR-121.

3. Fuel handling crane change. The applicant proposed to replace references to the fuel handling jib crane for the new-fuel handling crane. The basis for this change is addressed in TR-106.

4. The applicant proposed several changes throughout DCD Section 9.1.2 to reflect the spent fuel rack criticality analysis change addressed in TR-65, "Spent Fuel Storage Racks Criticality Analysis," APP-GW-GLR-029, Revision 3 of March 3, 2011 including resolving COL Information Item 9.1-4 by performing a confirmatory criticality analysis for the spent fuel racks.

9.1.2.2 Evaluation

The staff reviewed all changes identified in Section 9.1.2 of the AP1000 DCD. The staff did not re-review descriptions and evaluations of the spent fuel storage in AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in the applicant's TRs.

The regulatory basis for AP1000 DCD, Section 9.1.2, is documented in NUREG-1793. The staff has reviewed the proposed changes to DCD Section 9.1.2 against the applicable acceptance criteria of NUREG-0800 Sections 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" and 9.1.2, "New and Spent Fuel Storage." The following evaluations discuss the results of the staff's review.

The specific criterion that applies to the changes evaluated in this section is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.1.2.2.1 Spent Fuel Rack Design Change

TR-54, Revisions 0 and 1, described the design details and design-basis analyses for the spent fuel racks. To be consistent with the design of the spent fuel racks and the analyses presented in TR-54, Revisions 0 and 1, the following changes were proposed in DCD Revision 17, Section 9.1.2:

1. Delete references "to supporting grid structures at the top and bottom elevations" and replaced them with "a thick base plate at the bottom elevation" in DCD Section 9.1.2.1.

2. Increase SFP capacity from 619 fuel assembly locations to 889 fuel assembly locations.

3. Replace Figures 9.1-2 and 9.1-3, which show the rack array center-to-center spacing, with new Figures 9.1-2, 9.1-3, and 9.1-4, which show the SFP rack layout in Region 1, Region 2, and overall SFP rack layout, respectively.

4. Add the following statement in DCD Section 9.1.2.1: "All spent fuel racks will be in place whenever fuel is stored in the spent fuel racks. See DCD Section 3.7.5.2, for discussion

of site-specific procedures for activities following an earthquake. An activity will be to address measurement of the post-seismic event gaps between spent fuel racks and to take appropriate corrective actions."

5. Add the following statement in DCD Section 9.1.2.1: "The stress analysis of the spent fuel racks satisfies all of the applicable provisions in NRC RG 1.124, Revision 1 for components designed by the linear elastic analysis method."

6. Add design scope, in addition to SFP, the fuel transfer canal, and the cask loading pit, the FHM traverses the new fuel storage pit and the rail car bay in DCD Section 9.1.2.1.

7. Delete the following statement in DCD Section 9.1.2.2.1: "The purchase specification for the spent fuel storage racks requires the vendor to perform confirmatory dynamic and stress analyses."

8. Add the additional design information on the spent fuel racks center-to-center spacing; and material characteristics and recommended monitoring schedule of the neutron absorbing material (Metamic) coupons used in the fuel storage racks in DCD Section 9.1.2.2.1.

9. Add the following statement in DCD Section 9.1.2.2.1: "Both of these rack module configurations provide adequate separation between adjacent fuel assemblies with neutron absorbing material to maintain a subcritical array."

10. Change design activities: Reference to seismic and stress analyses of the spent fuel racks are changed to present tense (e.g., are performed) from future tense (e.g., will be performed by the vendor) in DCD Section 9.1.2.2.1.

11. Add item F in DCD Section 9.1.2.2.1, addressing loads due to "Internally Generated Missiles."

12. Changed Equipment Classification: In DCD Section 9.1.2.3, the racks are changed from Equipment Class 3 to Equipment Class D.

13. Delete the following statement in DCD Section 9.1.2.3: "because of the close spacing of the cells, it is impossible to insert a fuel assembly in other than design locations."

9.1.2.2.1.1 Spent Fuel Rack Structural

9.1.2.2.1.1.1 Summary of Technical Information

The staff confirmed that the proposed changes to DCD Revision 17, Section 9.1.2 are consistent with the design changes identified in TR-54, Revisions 0 and 1. In order to resolve the outstanding issues, and also to evaluate subsequent changes to the spent fuel rack design and changes to the spent fuel rack design basis, submitted to the staff in TR-54, Revisions 2 and 3. TR-54, Revision 4, was issued in June 2010, to update both the design and the design basis, and to address the unresolved RAIs.

In total, the staff issued 44 RAIs for TR-54. The significant RAIs are discussed below. Any RAIs not specifically discussed herein were for clarification only, and did not require a detailed technical response by the applicant.

9.1.2.2.1.1.2 Evaluation

Section 2.8.5 of TR-54 indicates that the three drop scenarios all assume a 91.44 cm (36 in) drop height above the top of the spent fuel storage rack. In RAI-TR54-01, the staff requested that the applicant describe the fuel handling operation that determines this drop height.

In a letter dated April 9, 2007, the applicant stated that the fuel handling operations for the Section 2.8.5 scenarios are the normal fuel handling operations such as those performed during refueling outages, i.e., core fuel offloaded and reloaded from/to the Reactor Building via the fuel transfer system into and out of the SFP storage racks. There are also instances where fuel inspection and/or fuel repair will require the fuel to be removed from the spent fuel storage racks and moved to the designated fuel inspection or fuel repair workstation. These fuel handling operations include the transfer of fuel from the rack cells and into the cask area during dry cask storage operations.

The applicant also stated that the current fuel transfer system in the spent fuel building, which lifts and lowers the fuel during normal fuel handling operations, consists of the FHM and the SFHT. The FHM is a fixed mast manipulator-type bridge crane similar in design to the manipulator crane bridges used in numerous existing Westinghouse operating plants' Reactor Buildings. The SFHT is a long handled tool which latches onto the fuel assembly top nozzle via a manually actuated gripper. Lifting of the SFHT and attached fuel assembly is performed using an auxiliary hoist on the FHM bridge. The design of the SFHT is very similar in design to the fuel handling tool currently in use at numerous existing Westinghouse operating plants.

The applicant indicated that the current designs of the AP1000 SFP, spent fuel storage racks, FHM, and SFHT limit the height that the fuel assembly can be lifted above the spent fuel racks to 22.86 cm (9 in) maximum. This height is limited by the water coverage above the fuel assembly and is limited by the physical design of the FHM and SPHT via mechanical stops and/or tool length. The maximum fuel drop height will be approximately 22.86 cm (9 in); which is bounded by the Section 2.8.5 scenarios of 91.44 cm (36 in).

At the June 2010 audit, the staff inquired about the operational controls on spent fuel handling that will ensure no exceedance of the analyzed 91.44 cm (36 in) drop height. In a letter dated July 13, 2010, the applicant submitted a supplemental response to RAI-TR54-01, describing how this will be accomplished. The staff reviewed the RAI response and concluded that there are sufficient hardware, software, and administrative controls in place to ensure that the design basis fuel assembly drop height of 91.44 cm (36 in) will not be exceeded. Therefore, RAI-TR54-01 is resolved.

As described in Section 2.8.5, the objective of the LSDYNA impact analyses is to assess the extent of the permanent damage to the rack and the structural integrity of the SFP liner. In RAI-TR54-02, the staff inquired whether the analyses are also intended to address the structural condition of the dropped fuel assemblies. If the analyses are also intended to address damage to the fuel assemblies, the staff would need additional fuel assembly design details and LSDYNA analysis details. If not, the applicant was asked to identify where this is addressed in the DCD or other technical report(s).

In a letter dated April 9, 2007, the applicant provided the following response:

> The LSDYNA impact analyses are not intended to address the structural condition of a dropped fuel assembly. The analysis addresses the structural condition of the rack and its ability to maintain subcriticality. A fuel handling accident and its radiological consequence is addressed in DCD Subsection 15.7.4 Fuel Handing Accident. The fuel handling accident described in Subsection 15.7.4 is defined as the dropping of a spent fuel assembly such that every rod in the dropped assembly has its cladding breached so that the activity in the fuel/cladding gap is released.

The applicant referenced report, APP-FS02-Z0C-001, Revision 0, “Analysis of AP1000 Fuel Storage Racks Subjected to Fuel Drop Accidents,” for the requested LSDYNA analysis details.

The staff confirmed that TR-54, Revision 1, removed any information describing the fuel assembly capacity and margins, since this is not the purpose of the TR-54. The staff also confirmed that TR-54, Revision 2, Section 2.8.1.4, and related sections in DCD Revision 17 were revised to include the maximum calculated fuel impact load on the racks (demand), and show it is less than the allowable impact load (capacity). Therefore, RAI-TR54-02 was resolved.

Section 2.8.5 of TR-54 indicates that a quarter of the spent fuel rack and a single fuel assembly were modeled appropriately using LSDYNA's shell and solid elements. The rack is submerged under the water when an impact occurs. In RAI-TR54-04, the staff requested the applicant confirm whether the water-structure interaction has been accounted for or to provide an explanation why this effect is not important.

In a letter dated April 9, 2007, the applicant stated that the fuel assembly drop analyses conservatively neglect the water structure interaction in the wake of an impact. By assuming that the impact occurs in air, as opposed to water, none of the impact energy is diverted to fluid kinetic energy (i.e., fluid damping), which would attenuate the deformation to the fuel rack. The applicant referenced report APP-FS02-Z0C-001, Revision 0.

The staff reviewed the applicant’s response and concluded that neglecting the mitigating effects of the water is conservative; therefore, the response is acceptable, and RAI-TR54-04 was resolved.

Section 2.8.5 of TR-54 states that appropriate non-linear material properties have been applied to the rack components to permit yielding and permanent deformation. TR-54 Table 2-6 provides Young's modulus, yield strength, and ultimate strength. LSDYNA requires a true stress-strain relation for nonlinear materials. In RAI-TR54-05, the staff requested the applicant to provide the following: (1) a complete description of the material stress-strain curve and confirmation that a true stress-strain curve was used in these impact analyses; and (2) a description of the fuel assembly model, including the element properties and material properties for the dropped fuel assembly.

In a letter dated April 9, 2007, the applicant provided the following response:

> The spent fuel racks are fabricated from SA240-304 and SA564-630 stainless steel. For the impact analyses, a true stress-strain curve, which is obtained from <u>Atlas of Stress Strain Curves</u> (2nd Edition, ASM International) and [included] as

> Figure TR-54-5.1 [in the RAI response], is used to define the strength properties of SA240-304 stainless steel.
>
> The properties of SA564-630, which is used to fabricate the adjustable support pedestals, are input in terms of engineering stress-strain, based on material data taken from the ASME Code. Also, the welds that connect the rack components are modeled as a bi-linear elasto-plastic material having the engineering stress-strain properties of the adjoining base metal (i.e., SA240-304). The material property values, which are used to define the engineering stress-strain curves for SA564-630 stainless steel and for the structural welds, are [summarized in the table included in the RAI response].
>
> The fuel assembly is modeled by a rigid bottom end fitting and a mass at the top (representing the weight of lifting tool) connected by an elastic beam (with a Young's modulus of 1.04 x 10^7 psi and a Poisson's ratio of 0.3 for typical rod material) that has an equivalent mass and total cross sectional area of all fuel rods in an AP1000 fuel assembly. In addition, a very thin rigid shell is attached to the bottom end fitting to represent the side surfaces of the fuel assembly that might be in contact with rack cell walls in a shallow drop event. To maximize the damage in the rack, the fuel assembly is only allowed to move in the vertical direction.

The staff reviewed the applicant's response, and determined that additional clarifications were needed, related to representation of nonlinear material properties and modeling of the fuel assembly for the drop analysis. During April 2007 and October 2007 regulatory audits, the staff discussed the needed clarifications with the applicant.

Since curves are only provided at 21.11 °Celsius (C) (70 °Fahrenheit (F)) and 430 °C (806 °F), the staff was not sure how the stress-strain curve at the required temperature (65.55 °C (150 °F)) was obtained from Figure TR- 54-5.1. The applicant indicated that the true stress-strain curve for the stainless steel material at the appropriate temperature (65.55 °C (150 °F)) is derived by manual interpolation of the true stress-strain curves at 21.11 °C (70 °F) and 430 °C (806 °F), which are provided in Atlas of Stress Strain Curves (2nd Edition, ASM International) for Type 304 stainless. The properties were linearly interpolated to obtain the values at 65.55 °C (150 °F). However, the staff requested the applicant to demonstrate whether using a linear interpolation approach is conservative. Using data from the ASME Code Section II, Part D, the applicant showed that the temperature versus yield stress and ultimate stress for stainless steel materials is not linear; the applicant's linear interpolation resulted in a slight overestimation of these values in the LSDYNA drop analyses. The overestimation was less than 4 percent for the ultimate strength and less than 10 percent for the yield. The applicant concluded that the results would not change significantly. The staff reviewed the differences in the two interpolated stress-strain curves, and concluded that the applicant's assessment that the results would not change significantly is acceptable.

The staff was uncertain of the effect that modeling the welds using bi-linear elasto-plastic material having engineering stress/strain properties, rather than the true stress-strain curve, has on the results of the analysis. For the weld, the applicant indicated that the use of the engineering stress strain curve is conservative because it is lower than the true stress strain curve and the failure strain is less. The staff found this acceptable.

From the response dated April 9, 2007, it was not clear to the staff whether using the given Young's modulus and total cross-sectional area of all the fuel rods provides an equivalent axial stiffness of the fuel pellet material and fuel cladding material. The applicant indicated that the axial stiffness of the dropped fuel assembly is based on the cladding material only, rather than the cladding and fuel pellets. The mass of the lifting tool is placed on top of this elastic beam. The mass of the cladding is included within the density of the beam. The fuel mass is lumped at the bottom of the fuel assembly model. The thin rigid shell surrounds the entire fuel assembly to position the single line beam representation within the storage cell.

The applicant agreed to expand the existing information for the fuel assembly model, in the next revision to TR-54. The staff verified that Section 2.8.5 of the TR-54, Revision 1, included sufficient information describing the fuel assembly model. Therefore, RAI-TR54-05 was resolved. The staff confirmed that TR-54, Revision 4, includes the same descriptive information.

The fuel rack baseplate shown in Figures 2-7 and 2-8 of TR-54 appears to have only one layer of 8 node brick element through its thickness. It is not clear if a solid or a thick shell element is used. In RAI-TR54-06, the staff requested the applicant to clarify the type of element used for the baseplate.

In a letter dated April 9, 2007, the applicant stated that the baseplate is modeled using 8-noded solid elements arranged in a single layer. The staff determined that the applicant would need to provide justification, demonstrating that modeling the baseplate using an 8-node solid element can adequately capture the behavior of the plate, including bending through the thickness. The staff noted that the model is used for the drop analyses and the seismic analyses of the racks.

During the April 2007 audit, the staff discussed this concern with the applicant. The applicant agreed to provide a supplemental response to address the staff's concern.

During the October 2007 audit, the applicant indicated that the rack baseplate model was revised to utilize thick shell elements in HOLTEC Report No. HI-2063519, Revision 1. The applicant also indicated that they revised their model to use strain rate effects for the material properties. The net effect of both improvements resulted in lower deformations. The staff found the use of the thick shell element representation of the baseplate to be appropriate because it more closely simulates the true behavior under dynamic impact loadings.

During the May 2008 regulatory audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows the rack baseplate modeled with thick shell elements, which can capture the bending behavior of the plates. In addition, strain rate effects were also included in the spent fuel rack analysis. These new analyses demonstrate that the maximum downward deformation of the rack baseplate is 7.98 cm (3.14 in), which is less than the 8.89 cm (3.5 in) used in the criticality analysis. Also, the criticality analysis was conservatively performed assuming a total of nine fuel assemblies deform the maximum value of 8.89 cm (3.5 in) rather than the single drop at 7.98 cm (3.14 in). Therefore, RAI-TR54-06 was resolved.

Section 2.8.5 of TR-54 indicates that the baseplate of the rack is connected to the cells by appropriate welding. However, the cells are described in Sections 2.1.1.1 and 2.1.1.2 as resting on top of the baseplate. Welded connections between the cells and the baseplate would greatly increase the strength of the whole rack system. The staff issued RAI-TR54-07, which reads as follows:

a) Confirm there is a welded connection between the baseplate and the cells.
b) Describe the design details of this connection.
c) Describe how this connection is modeled in LS-DYNA.

In a letter dated April 9, 2007, the applicant provided the following response:

a) The base of every storage cell is welded to the rack baseplate.

b) Each cell is welded to the baseplate on four sides by 1/16 in fillet welds having a minimum length of 7 ".

c) The cell to baseplate weld connection is modeled in LSDYNA by shell elements, which join the bottom of the cell and the baseplate top surface, with a thickness equal to the corresponding throat dimension of the weld.

The staff determined that the applicant provided sufficient information to clarify the cell-to-baseplate attachment, and found the response acceptable. Therefore, RAI-TR54-07 was resolved.

For the drop case in which the impact occurs directly above a rack pedestal, Section 2.8.5 of TR-54 provides the concrete strength of the pool floor and the thickness of the stainless steel liner, but does not provide the thickness of the pool floor. There is a possibility that the impact could also cause damage to the concrete floor, and pose a more severe consequence than causing the liner to yield. The maximum Von Mises stress in the SFP liner is reported as 161.3 MPa (23.4 kips per square inch (ksi)), which is much larger than the concrete strength of 27.6 MPa (4 ksi); the concrete may crush and crack locally at this level of stress. In RAI-TR54-08, the staff requested the applicant to provide additional details on the modeling of the concrete floor (including a figure of the concrete model, element type, boundary conditions, material properties, etc.) and the analysis results for the concrete floor (in addition to Figure 2-11).

In a letter dated April 10, 2007, the applicant stated:

> The spent fuel pool concrete floor is modeled only in the vicinity of the impacted rack pedestal with an assumed thickness of two ft and compressive strength of 4,000 psi. The pool liner and rack pedestal bearing pad are also modeled as shown in Figure TR54-8.1 [attached to the RAI response]. The periphery surface nodes of the SFP pool liner and the underlying concrete slab in the LSDYNA model are restrained from moving in the vertical direction and in the horizontal direction normal to the periphery surface to simulate the confining effect of the surrounding structure.
>
> The maximum compressive stress in the concrete floor, resulting from the fuel assembly deep drop event in which the impact occurs directly above a rack pedestal, is predicted to be 4,557 psi as shown in the Figure TR-54-8.2 [attached to the RAI response]. This maximum compressive stress slightly exceeds the assumed concrete compressive strength and is limited to the top surface of the concrete near the bearing pad edge. The very limited local damage to the concrete floor surface is acceptable since the acceptance criterion for the fuel deep drop accident is no gross failure of the SFP floor leading to an uncontrolled loss of SFP water.

The staff determined that the response is acceptable because the bearing capacity of confined concrete is greater than the compressive strength of 27.6 MPa (4,000 psi). As follow-up, the staff requested the applicant to describe the concrete material model used in LSDYNA. During the April 2007 audit, the applicant indicated that concrete nonlinear material model “Material 16" was used in the LSDYNA analysis. The staff requested the applicant to explain how the input parameters for Material 16 were developed and verified, since they do not appear to be derivable from commonly known material properties of concrete.

During the October 2007 audit, the applicant indicated that the properties of Material 16 in LS-DYNA were obtained from NUREG/CR-6608, “Summary and Evaluation of Low Velocity Impact Tests of Solid Steel Billet into Concrete Pads,” (February 1998). This concrete model was used by HOLTEC in the generic license for HI-STORM 100 storage casks, which received approval by the NRC. The staff requested the applicant confirm that the 27.6 MPa (4,000 psi) concrete compressive strength for the AP1000 fuel pool structure is consistent with the concrete strength used in the referenced NUREG/CR-6608 report. The applicant stated that the concrete pad modeled in NUREG/CR-6608 has a compressive strength of 29.0 MPa (4,200 psi), which is slightly greater than the compressive strength of the AP1000 SFP slab (27,6 MPa (4,000 psi)). In order to account for the difference in strength, the input parameters specified in Appendix C of NUREG/CR-6608, for use with LS DYNA Material Model 16, have been modified in accordance with HOLTEC Position Paper DS-240 for a compressive strength of 27.6 MPa (4,000 psi).

Since the concrete input parameters were adjusted to account for the differences in compressive strength between the concrete used in NUREG/CR-6608 and the concrete used in the AP1000 design, and considering the small difference in compressive strengths, the staff found this acceptable. Therefore, RAI-TR54-08 was resolved.

Section 2.8.5 of TR-54 does not indicate whether other fuel assemblies are in place when a fuel assembly drops through an empty cell and impacts the baseplate at its center. Depending on how the baseplate is designed, a full load of fuel assemblies may introduce progressive deformation after a fuel assembly impacts at the center of the baseplate. The maximum downward deformation of the baseplate is about 10.92 cm (4.3 in), as shown in TR-54 Figure 2-10. This may be significant enough to initiate a progressive deformation. Therefore, in RAI-TR54-09, the staff requested the applicant to provide: (1) the assumption for in-place fuel assemblies when the impact occurs; (2) the design basis for the baseplate; and (3) a figure similar to Figure 2-10 that shows the cells together with the severely deformed baseplate.

In a letter dated April 10, 2007, the applicant provided the following response:

(1) The spent fuel storage rack is assumed to be empty (i.e., no fuel assemblies in place) when a fuel assembly drops through an empty cell and impacts the baseplate at its center. This is a simplifying assumption, which is reasonable considering that the buoyant weight of a fuel assembly is approximately 1,525 lb whereas the impact load transmitted by the dropped fuel assembly is roughly 268,000 lb based on the LSDYNA solution.

(2) The design basis for the baseplate is to provide vertical support for the stored fuel assemblies and to protect the Spent Fuel Pool liner from a fuel assembly strike. In other words, a dropped fuel assembly should not pierce the baseplate and result in a direct impact with the liner.

During the April 2007 audit, the staff discussed the response with the applicant and requested a supplemental response for items (1) and (3). For item (1) the applicant indicated they would provide a supplement to the response, to address the effects of the additional fuel assemblies during the vertical drop accidents. For item (3) the applicant agreed to provide another detail showing the bottom plate and cell walls in the surrounding region where the maximum deformation and stresses occur.

In a letter dated May 17, 2007, the applicant submitted the revised RAI response. The applicant's response to items (1) and (2) was unchanged. For item (3), the applicant attached a figure to the response, showing the cells together with the severely deformed baseplate, for the same LS DYNA solution as shown in Figure 2-10 of TR-54. The applicant noted that the deformation of the cells is not significant compared to the baseplate, because the cell to baseplate weld connections broke as a result of the postulated fuel impact load before the cell walls were permanently deformed. The staff reviewed the applicant's revised response, and found the response to item (3) acceptable.

During the October 2007 audit, in response to item (1), the applicant indicated that the rack model, represented in HOLTEC Report No. HI-2063519, Revision 1, had been revised to consider the effects of all of the stored fuel assemblies in the rack, by modifying the density of the rack baseplate. The modeling of the baseplate was also changed to thick shell elements, and strain rate effects were included. The staff reviewed the HOLTEC calculation and confirmed that it includes the mass effect of all of the fuel assemblies by increasing the baseplate density. The staff concluded that the consideration of the rest of the fuel assemblies by increasing the mass of the baseplate is an acceptable approach to simulate the dynamic effects. The staff requested the applicant to finalize the HOLTEC calculation and revise TR-54, to describe the modeling approach used. The staff noted that its concern about the large vertical deformation (10.92 cm (4.3 in)) is addressed under RAI-TR54-10.

During the May 2008 audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows that the rack is conservatively assumed to be fully loaded with fuel assemblies, except at the center cell where the dropped fuel hits the rack baseplate. Therefore, RAI-TR54-09 was resolved.

In Section 2.8.5 of TR-54, a vertical movement of 5.08 cm (2 in) of a fuel assembly is defined as the criticality limit, and the impact analysis shows that quite a number of fuel assemblies will have more than 5.08 cm (2 in) displacement. It appears that a rack design with only a 5.08 cm (2 in) space between the bottom of the baseplate and the top of the floor would eliminate this risk. In RAI-TR54-10, the staff requested the applicant to explain why the design has a space larger than 5.08 cm (2 in).

In a letter dated April 9, 2007, the applicant stated that each spent fuel rack storage cell is 506.73 ± 0.159 cm (199.5 ?± 0.0625) in) in length and rests on top of a base plate whose top is 5 in above the SFP liner. Each Metamic poison panel is 436.9 cm (172 in) long and has a bottom elevation that is 15.8 cm (6.23 in) above the top of the base plate. The active fuel region of each fuel assembly begins at an elevation 20.9 cm (8.23 in) above the top of the base plate. Therefore, the bottom elevation of the Metamic poison panel is positioned to be 5.08 cm (2 in) lower than the bottom elevation of the active fuel.

Therefore, the results of the criticality analyses are bounding even if the fuel assembly is vertically displaced downward by up to 5.08 cm (2 in) as a result of the hypothetical fuel assembly drop. The 5.08 cm (2 in) vertical displacement of the fuel assemblies, mentioned above in RAI-TR54-10, is not a criticality limit. The criticality analyses summarized in TR-65, Revision 0, addressed the hypothetical fuel assembly drop in Section 2.4.6.3 as follows:

> A fuel assembly (with a control rod and attached to the fuel assembly handling tool) is dropped and impahcts the baseplate as discussed in Subsection 2.8.5 of Reference 4. The analysis in Subsection 2.8.5 of Reference 4 indicates that the base plate will be defomed as a result of this drop. For conservatism, the fuel assemblies in nine storage cell locations will be modeled as vertically dropped by 3.5 inches to bound the consequences of the base plate deformation.

Since the criticality analysis demonstrates that the stored fuel assemblies remain subcritical following a hypothetical fuel assembly drop, the space between the bottom of the baseplate and the top of the floor is not designed to control criticality, but to protect the SFP liner from an impact strike. In other words, the rack baseplate is raised high enough above the floor (10.8 cm (4.25 in)) to prevent the baseplate from contacting the SFP liner when the baseplate deforms under impact.

The staff reviewed the above response and concluded that the response is acceptable. However, it should be noted that the maximum displacement of 8.89 cm (3.5 in) reported here does not agree with the 10.9 cm (4.3 in) indicated in Figure 2-10 of TR-54. During the April 2007 audit, the staff noted that the applicant needed to address the higher base plate deformation (10.9 cm (4.3 in)) in the vertical direction and how this may affect the criticality analysis.

During the October 2007 audit, the applicant indicated that the hypothetical drop, wherein a fuel assembly travels downward through an empty storage cell and impacts the baseplate, has been re-analyzed in HOLTEC Report No. HI-2063519, Revision 1, for both the Region 1 and Region 2 spent fuel racks. The new analysis model incorporates the following changes (as discussed in the RAI responses to RAI-TR54-06, RAI-TR54-09, and RAI-TR54-11): (1) the baseplate is modeled with thick shell elements; (2) the effect of the stored fuel assemblies is accounted for by increasing the mass density of the baseplate; and (3) strain rate effects are considered for the welds only. Based on the new analyses for the Region 1 and Region 2 spent fuel racks, the maximum vertical displacement of the rack baseplate is 7.98 cm (3.14 in), which is less than the 8.89 cm (3.5 in) displacement considered in the criticality analysis. Therefore, the existing criticality analysis remains bounding.

The improvements made to the fuel rack models were reviewed by the staff and found to be technically acceptable. The staff requested the applicant to finalize the HOLTEC calculation and to revise TR-54 to describe the modeling approach used.

During the May 2008 audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows that: (1) the baseplate is modeled with thick shell elements; (2) the effect of the stored fuel assemblies is accounted for by increasing the mass density of the baseplate; and (3) strain rate effects are considered for the welds. The resulting vertical deformation due to the drop is 7.98 cm (3.14 in). Therefore, RAI-TR54-10 was resolved.

Figure 2-9 of TR-54 shows the permanent deformation at the top of a cell wall at Region 2. The permanent deformation is measured as 50.8 cm (20 in), which is just slightly smaller than the limit of 52.07 cm (20.5 in). Since the deformation at the impact location is so close to the limit (i.e., very little margin exists), the mesh should be locally refined, to ensure convergence with mesh size. Therefore, in RAI-TR54-11, the staff requested the applicant to perform an additional analysis with a finer mesh at the impact region to confirm that the LSDYNA model is suitable.

In a letter dated April 9, 2007, the applicant stated that the general acceptance criterion for the 91.44 cm (36 in) fuel assembly drop onto the top of a Region 2 rack is to maintain the stored fuel assemblies in a subcritical configuration. In measurable terms, the permanent deformation of the rack (measured downward from the top of rack) is limited to 52.07 cm (20.5 in), which is equal to the distance from the top of the rack to the top of the neutron absorber panel. This limit is conservative because the active fuel region begins 5.08 cm (2 in) below the top of the neutron absorber panels. Therefore, more margins exist than TR-54 indicates, and a mesh convergence study is not required.

The staff noted that, although the distance from the top of the rack to the top of the fuel is 57.15 cm (22.5 in), which gives a margin of 6.35 cm (2.5 in), this is still a relatively small margin. Because of the small margin, the staff requested the applicant to demonstrate whether the finite element model is sufficiently refined in the impact region.

During the April 2007 audit, the staff discussed this with the applicant. The applicant noted that the neutron absorber panels, beyond the 50.8 cm (20 in) deformation, respond in the elastic range and would not be damaged. The applicant also noted that the criticality analysis was performed for 9 fuel assemblies having the active fuel region exposed 8.89 cm (3.5 in) (see response to RAI-TR54-10).

During the October 2007 audit, the applicant indicated that the 91.44 cm (36 in) fuel assembly drop onto the top of a Region 2 rack has been re-analyzed in HOLTEC Report No.HI-2063519, Revision 1, with consideration of strain rate effects for the welds. The new analysis shows that the maximum permanent deformation of the rack cell wall is only 35.7 cm (14.06 in) (measured from the top of rack) versus the allowable limit of 53.71 cm (21.145 in) as defined in the analysis report. Since the margin between the calculated deformation and the allowable limit is greater than 17.78 cm (7 in), the applicant stated that there is no longer a need to demonstrate that the refinement of the model is adequate in the localized region of the impact zone.

The staff, however, requested the applicant to confirm the adequacy of the rack model in the crushed zone region by providing curves that compare the hourglass energy to the kinetic, internal, and/or total energy. The applicant provided these curves, which showed that the hourglass energy was essentially negligible in comparison to the internal energy of the cell structure and impact bar that were being plastically deformed during these drop accident cases. In view of the now much larger margins in the extent of plastic deformation in the new revised model, and the comparison of the hour glass energy, the staff found the response technically acceptable, pending the finalization of the HOLTEC calculation and revision of TR-54 to describe the new results.

During May, 2008 audit, the staff verified that HOLTEC Report No. HI-2063519, Revision 2, shows the results of the drop analyses onto the top of the fuel racks considering the strain rate effects for the welds. In view of the now much larger margins in the extent of plastic deformation in the new revised model, and the comparison of the hour glass energy, the staff found the response technically acceptable. The TR does not go to this level of detail; however, the staff confirmed the HOLTEC calculation incorporated the above modeling change. Therefore, RAI-TR54-11 was resolved.

There are a total of six impact analyses for the Region 1 and Region 2 racks (3 drop cases for each rack region). TR-54 only presents the results for three analyses, on the basis that these are the bounding conditions. In RAI-TR54-13, the staff requested the applicant to explain the technical basis for concluding that these are the bounding conditions, or provide the results for the three analyses not presented in the report.

In a letter dated May 17, 2007, the applicant stated that the analysis was performed for both the Region 1 and Region 2 racks. The bounding analysis was reported. The shallow drop event involving a Region 1 rack was analyzed and found to yield a plastic deformation of 38.1 cm (15.0 in) measured vertically from the rack top, which is bounded by the reported Region 2 rack shallow drop analysis with a predicted rack top plastic deformation of 50.8 cm (20 in). Similarly, Region 2 rack baseplate was found to deform less than 8.38 cm (3.3 in), which is smaller than the reported bounding baseplate deformation of 10.83 cm (4.264 in) for the Region 1 rack. Finally, because the Region 1 rack is lighter than the Region 2 rack, more impact energy can be transferred into the SFP floor and therefore results in bounding SFP floor damage in an event where the fuel assembly drops directly over a Region 1 rack pedestal; this bounding case was analyzed and reported.

The staff evaluated the RAI response and determined that the response provided an adequate explanation of why the three accident drop cases that were analyzed bound the other three drop cases. Therefore, the staff found the response acceptable and RAI-TR54-13 was resolved.

In accordance with NUREG-0800 Section 3.8.4, Appendix D, one of the fuel handling accident loads that need to be considered is uplift force on the rack caused by a postulated stuck fuel assembly. Section 2.8.3 of TR-54, Revision 0, states: "An evaluation of a stuck fuel assembly, leading to an upward load of 2,000 lb has been performed. The results from the evaluation show that this is not a bounding condition because the local stresses do not exceed 2,500 psi." The staff determined that additional information was needed in order to assess whether this load has been adequately considered. In RAI-TR54-14, the staff requested that the applicant provide a detailed description of the assumptions, the analyses conducted, the results obtained, and the basis for the conclusion that this is not a bounding condition.

In a letter dated May 17, 2007, the applicant stated: “A nearly empty rack with one corner cell occupied is subject to an upward load of 2000 lbf, which is assumed to be caused by the fuel sticking while being removed." The applicant attached a calculation, and stated that the local stress is well below the yield stress of the cell wall material (i.e., 147 MPa (21,300 psi) pursuant to Table 2-6). The applicant also noted that the value of 17.2 MPa (2,500 psi) will be changed in Section 2.8.3 to 20.7 MPa (3,000 psi) for the local stresses resulting from a stuck fuel assembly.

In a letter dated April 18, 2008, the applicant augmented its initial RAI response, stating that the calculation provided in the RAI response is excerpted from HOLTEC Report No. HI-2063523, (APP-FS02-S3C-001, Revision 0). The staff evaluated the supplemental RAI response, including the calculation that demonstrates the adequacy of the vertical welds along the height between adjoining cells and the horizontal welds at the base (cell walls to baseplate). The staff found all responses to staff’s concerns are satisfactory, pending revision to the HOLTEC calculation and the TR.

In December 2008, the staff reviewed TR-54, Revision 2, and related sections in DCD Revision 17. The staff noted that a new Section 2.8.6 has been added to TR-54, Revision 2, to describe the stuck fuel assembly evaluation. Section 2.8.6 states, “A nearly empty rack with one corner cell occupied is subject to an upward load of 5,000 lbf, which is assumed to be caused by the fuel sticking while being removed." This increase in uplift force was necessitated by changes in the spent fuel handling operations. The staff found the re-analysis to be acceptable. The predicted stresses are still below allowable stresses. Therefore, RAI-TR54-14 was resolved.

The staff noted that the descriptive information included in TR-54 was not sufficient to permit an adequate review of the structural/seismic analysis of the spent fuel racks, in accordance with NUREG-0800 Section 3.8.4, Appendix D. In RAI-TR54-15, the staff requested the applicant provide descriptive information, including plans and sections showing the spent fuel racks, pool walls, liner, and concrete walls. All of the major features of the racks including the cell walls, baseplate, pedestals, bearing pads, neutron absorber sheathing, any impact bars, welds connecting these parts, and any other elements in the load path of the racks should be shown on one or several sketches. These sketches should also indicate related information which includes key: cutouts, dimensions, material thicknesses, and gaps (fuel to cell, rack to rack, rack to walls, and rack to equipment area). In addition to the above, for review of postulated fuel handling drop accident and quantification of the drop parameters, sketches with sufficient details for the fuel handling system should be provided to facilitate the review as indicated in NUREG-0800 Section 3.8.4, Appendix D.

In a letter dated June 8, 2007, the applicant provided sketches in the attachment to the RAI-TR54-15 response, showing the major features of the racks and SFP. The applicant stated that these sketches would be incorporated in TR-54. The detailed design of fuel handling equipment and detailed sketches are not available. However, the quantification of the drop parameters has been established in the DCD (both maximum drop weights and heights). The DCD drop heights are much greater than what is being designed for the fuel handling equipment, which is stated in the RAI-TR54-01 response.

During the April 2007 audit, the complete design drawings of the spent and new fuel racks were available to the NRC for review. In addition, HOLTEC explained how the rack features were incorporated into the seismic/structural models.

During the October 2007 audit, the staff discussed five specific items with the applicant: key dimensions of the male and female pedestal components and bearing plates; welds connecting the pedestals to the baseplate; welds connecting the baseplate to the fuel cell walls; leak chase channels; and gaps between the racks. The applicant provided additional information to the staff, describing these details.

During the May 2008 audit, the staff reviewed all the pending revisions to TR-54 and the DCD. The staff identified the need for additional changes to certain figures in TR-54 and the need to include them in the DCD for consistency.

In a letter dated June 20, 2008, the applicant provided its supplemental response to this RAI. The staff reviewed the response and found that TR-54, Revision 2, and related sections in DCD Revision 17 had been appropriately revised. Therefore, RAI-TR54-15 was resolved.

The staff noted that TR-54 is a summary report. However, to adequately perform a technical review of the analysis and design of the spent fuel racks, a more detailed report should have been submitted, similar to those provided in past technical reviews of spent fuel racks for specific nuclear power plants. Therefore, in RAI-TR54-16, the staff requested the applicant provide the detailed spent fuel storage rack report/calculation for review.

In a letter dated April 9, 2007, the applicant stated that TR-54 is a summary report; and two calculations [HOLTEC Report No. HI-2063523, “Spent Fuel Rack Structural/Seismic Analysis for Westinghouse AP1000,” Revision 0, 08/15/2006 (APP-FS02-S3C-001); and HOLTEC Report No. HI-2063519, “Analyses of AP1000 Fuel Storage Racks Subjected to Fuel Drop Accidents,” Revision 0, 08/15/2006 (APP-FS02-Z0C-001)] form the basis for TR-54.

During the April 2007 audit, the staff reviewed both HOLTEC calculations. Based on this review, the staff requested the applicant to address nine specific questions related to gaps between racks, modeling assumptions, and solution convergence in the seismic analysis. Two significant issues required the applicant to conduct additional analysis. The staff requested that the applicant provide a justification for the assumptions used in developing the various spring stiffnesses, including how variations in the spring stiffnesses affect the results (e.g., sensitivity studies). The staff also requested the applicant to address the sensitivity of the numerical results to the integration time step used in the analysis. The remaining seven questions were requests for clarifications, not requiring additional analysis. The staff also requested the applicant to address 11 specific questions related to design parameters (e.g., weight, temperature, flow) and to analysis methods used for the drop accident analysis. These questions were requests for clarifications, not requiring additional analysis. The applicant agreed to address all of the staff’s questions by making appropriate changes and/or additions to the calculations.

During the October 2007 audit, the applicant indicated that, in order to address possible variations in the spring constants, a sensitivity study was performed in which the calculated impact spring constants were uniformly increased and decreased by 20 percent in two separate computer runs. The COF used for all three computer runs (i.e., 80 percent, 100 percent, and 120 percent of the calculated spring constants) is 0.8. The applicant also indicated that the time integration step used for the computer runs is 1×10^{-5} sec. In order to verify that the runs are converged, an additional computer run (Run 6) was performed using a time integration step of 5×10^{-6} sec (i.e., half of the original time step). The differences in results are minimal. HOLTEC Calculation HI-2063523 was updated to include the results of these sensitivity studies, and the final safety conclusions are based on the maximum results from all computer runs, including the sensitivity studies. The staff reviewed HOLTEC Calculation HI-2063523, Revision 1, and noted that the calculation had been revised to include the numerical results for the sensitivity studies, and that these results were considered with all the other runs in the structural assessments.

During the August 2009 audit, the staff confirmed that all 20 of the staff’s questions had been appropriately addressed in updates to the HOLTEC calculations and corresponding Westinghouse reports. Therefore, RAI-TR54-16 was resolved.

The staff noted that insufficient data were provided in TR-54 describing the seismic input loads used for analysis of the spent fuel racks. The staff issued RAI-TR54-17, which reads as follows:

a. Floor response spectra (X, Y, and Z - vertical directions) at or the near the elevation of the top of the fuel racks and near the bottom of the fuel rack or pool floor corresponding to the damping value used for the analysis.

b. Explain why the envelope of these two sets of spectra was not used.

c. The current DCD is applicable for the hard rock site. Therefore, provide further explanation for the range of soil and rock properties used in enveloping the seismic floor spectra. Where are these ranges of soil/rock properties specified for confirmation by a future COL applicant?

d. For the synthetic time histories, provide plots of the three time histories, the cross correlation coefficients, the comparisons of the spectra from the synthetic time histories to the enveloped target response spectra, and the comparisons of the power spectral density plots to the target power spectral density function associated with the target response spectra.

e. Which time history was used (displacement, velocity, or acceleration)? Were all three directions input simultaneously? Was gravity included in the time history analysis?

In a letter dated April 9, 2007, the applicant provided the following response:

a. Floor response spectra (X, Y, and Z vertical directions) near the elevation of the bottom of the spent fuel pool floor corresponding to the damping value used for the analysis are provided in the PDF attachment RAI-TR54-17a. No Floor response spectra are provided near or at the elevation of the top of the spent fuel racks (See response to RAI-TR54-17b).

   The ASB99 floor response spectra represent the enveloping response spectra for the auxiliary shield building at elevation 99 ft for a range of soil/rock condition. FRS of various soil/rock analyses were first enveloped for various locations of the ASB. All of the ASB locations at elevation 99 ft were then grouped and enveloped to develop the ASB99 floor response spectra. The spent fuel pool is at a lower elevation but the dynamic response is essentially the same as at elevation 99 ft.

b. The spent fuel racks are free-standing in the spent fuel pool. They are not anchored to the spent fuel pool walls. The spent fuel racks are excited in a seismic event by the floor response spectra representing the spent fuel pool floor (ASB99). There is no need to envelope multiple sets of floor response spectra.

c. The range of soil and rock conditions for which the seismic floor spectra applies is described in Westinghouse technical report “Extension of NI Structures Seismic Analysis to Soil Sites.”

d. The synthetic time histories, the response spectrum curves, and the power spectral density plots for the Auxiliary Shield Building at Elevation 99 ft are provided in Figures TR54-17.1 through TR54-17.9 (attached to RAI response). The cross correlation coefficients for the three orthogonal components (East West, North South, and Vertical) of the ASB99 synthetic time histories are summarized in the table [submitted as part of this response].

e. Acceleration time histories are used as the input motion for the seismic analysis of the spent fuel racks. The acceleration input is defined by three orthogonal components, which are input and solved simultaneously. Gravity is also included in the time history analysis.

The staff found this RAI response to be acceptable because it adequately addressed all of the staff's questions. RAI-TR54-17 was initially resolved. However, subsequent to the initial resolution of this RAI, the applicant revised the seismic design loads twice. Therefore, during the June 2010 audit, the staff requested that the applicant update this RAI response to reflect the current seismic design loads for the spent fuel racks. In a letter dated July 9, 2010, the applicant submitted a revised response to RAI-TR54-17, updating the seismic design loads. The staff finds that the revised RAI response adequately describes the current seismic design loads for the spent fuel racks. TR-54, Revision 4, includes the numerical results for the current design loads. Therefore, RAI-TR54-17 is resolved.

The staff noted that the seismic analyses only considered the bounding values (0.2 and 0.8) for the COF between the pedestal and the pool liner. In RAI-TR54-18, the staff requested the applicant to provide the technical basis for only considering these two bounding values and not other intermediate values. The staff also requested clarifications about sliding. What is assumed to slide, the pedestal to bearing plate or bearing plate to pool liner? If it is the surface between the bearing plate and pool liner, how is damage to the pool liner due to horizontal forces avoided? Are there any physical provisions to prevent the bearing plate and pedestal from sliding to the point that the pedestal centerline would be at or beyond the edge of the bearing plate?

In a letter dated April 10, 2007, the applicant stated that the coefficients of friction used in the seismic analyses, namely 0.2 and 0.8, are consistent with previous spent fuel rack license applications, and they are based on the experiments performed by E. Rabinowicz (TR-54, Reference 21: "Friction Coefficients of Water Lubricated Stainless Steels for a Spent Fuel Rack Facility," MIT, a report for Boston Edison Company, 1976). The lower value of 0.2 produces the maximum sliding displacement between the rack pedestals and the bearing plates. The higher value of 0.8 increases the rocking motion of the spent fuel racks and produces the maximum stress in the rack pedestals. Sliding occurs between the rack pedestal and the bearing plate since these two items are made of different materials (SA564-630 vs. SA240-304), whereas the pool liner and the bearing plate are made of the same material (SA240-304) and are more likely to gall. There are no physical provisions to prevent the rack pedestal from sliding beyond the edge of the bearing plate. The seismic analyses, however, demonstrate that the maximum sliding displacement at the base of the rack is less than the distance between the pedestal outside diameter and the edge of the bearing plate.

The staff reviewed the applicant's response and noted that three COF values (0.2, 0.5, and 0.8) are used for the new fuel rack analysis. The applicant needed to justify why all three values are not utilized for the spent fuel rack analysis. Consideration of an intermediate value is appropriate because the analyses are highly nonlinear and it is not evident which value(s) would govern. The need to consider other values is also discussed in Section 6.4.2 of NUREG/CR-5912, "Review of the Technical Basis and Verification of Current Analysis Methods Used to Predict Seismic Response of Spent Fuel Storage Racks." The staff also requested the applicant to provide the maximum horizontal displacement from the analyses and compare that against the distance between the pedestal centerline and the edge of the bearing plate.

During the October 2007 audit, the applicant indicated that an additional computer run had been performed for an intermediate COF of 0.5. During the May 2008 audit, the staff reviewed HOLTEC Calculation HI-2063523, Revision 1, and confirmed that the report describes the analysis and results for the additional case of a 0.5 COF. In addition, the staff reviewed TR-54, Revision 1, and confirmed that the 0.5 COF case was included. Therefore, RAI-TR54-18 was resolved.

The staff noted that the SFP is divided into 2 regions, with different rack designs. In RAI-TR54-20, the staff requested the applicant to explain the reason for the different type racks (i.e., Region 1 and Region 2). If it is because of different fuel assembly types, explain how the analysis considers the various types and combinations of fuel assemblies (e.g., mass, sizes, gaps, fluid coupling, etc.).

In a letter dated April 9, 2007, the applicant stated that the AP1000 uses only one fuel assembly type. The purpose of the Region 1 racks is to provide storage for up to 243 fresh fuel assemblies with a maximum initial enrichment up to 5.0 w% U-235. This is accomplished by spacing these storage cells on a pitch equal to 27.67 cm (10.9 in.) and employing a “flux trap” poison configuration between consecutive storage cells.

The purpose of the Region 2 storage racks is to provide storage for up to 646 fuel assemblies in a high density configuration. These storage cells employ a pitch equal to 22.93 cm (9.028 in.) and a single poison panel separates consecutive fuel assemblies.

The response clarified that all of the fuel assemblies are the same; only the rack configuration is different between Region 1 and Region 2. Since all of the racks are included in the 3-dimensional pool rack model, the staff found the response acceptable. Therefore, RAI-TR54-20 was resolved.

In RAI-TR54-21, the staff requested the applicant to explain how the different impact stiffness values are determined for the fuel to cell wall, rack to rack, rack to wall, and pedestal to floor. Since the impact forces are affected by the impact spring constants, how is the sensitivity of the impact forces and rack responses to variation in these spring constants addressed? Are impact forces imparted directly onto the cell walls or are there impact bars?

In a letter dated May 17, 2007, the applicant stated that the impact stiffness values for the rack to rack, rack to wall, and pedestal to floor are calculated as shown in the attachment to the RAI response. The fuel to cell wall impact stiffness is determined based on the solution for a simply supported circular plate under a concentrated load applied at its center, where the plate diameter is equal to the cell inner dimension and the plate thickness is equal to the cell wall thickness. The stiffness of the annular plate is then multiplied by the number of loaded storage cells for each rack, since the stored fuel assemblies are assumed to rattle in unison. A sensitivity study has not been performed specifically for the AP1000 spent fuel racks to quantify the effect of variations in the impact stiffness values. However, sensitivity studies have been performed in the past for similar spent fuel rack applications submitted by HOLTEC, which employed the same method of computing the impact stiffness values, and the impact forces were found to be insensitive to small variations in the stiffness values provided that the integration time step was sufficiently small. There are impact bars around the entire perimeter of each Region 2 spent fuel rack at the top of the rack. These bars prevent impact forces from being imparted directly onto the cell walls, and they reinforce the rack cell structure at the point of impact.

During the October 2007 audit, the staff discussed with the applicant the development of the spring constant for impact loads between the stored fuel assemblies and the cell walls, which is based on the solution for a circular plate with simply supported edges subjected to a uniform pressure load. This approach has been used consistently by HOLTEC since the mid 1980’s, when the computer code DYNARACK was first developed. As a result, numerous spent fuel rack licensing applications over the past 20 years have relied on this approach. The applicant also noted that, in response to RAI-TR54-16, a sensitivity analysis was performed in which all spring constants used in the DYNARACK model were uniformly increased and decreased by 20 percent. The stiffer springs resulted in only a 0.4 percent increase in the fuel-to-cell impact load.

The staff concluded that the applicant adequately addressed the sensitivity of the response to variations in the spring constants. Therefore, RAI-TR54-21 was resolved.

Section 2.2.2.2 of TR-54 describes some modeling information for a single rack. It indicates that the rack cellular structure is modeled by a 3-D beam having 3 translational and 3 rotational DOFs at each end, so that two-plane bending, tension/compression, and twist of the rack are accommodated. In RAI-TR54-23, the staff requested the applicant to explain why shear stiffness/deformation is not also included, and to provide more detailed information about how the beam model of the rack was developed, considering that it is an assembly of many square-celled structures welded at discrete locations.

In a letter dated April 9, 2007, the applicant stated that the shear deformation is included in the rack dynamic model. The beam model of the rack was developed based on the applicable Codes, Standards and Specifications given in Section IV(2) of the NRC guidance on SFP modifications entitled, “Review and Acceptance of Spent Fuel Storage and Handling Applications,” dated April 14, 1978, which states that “Design … may be performed based upon the AISC specification or Subsection NF requirements of Section III of the ASME Code for Class 3 component supports.” The rack modeling technique is consistent with the linear support beam element type members covered by these codes.

The staff verified that Section 2.2.2.2 of the TR-54, Revision 1, was revised to clarify that shear stiffness/deformation is also included in the rack model. Therefore, RAI-TR54-23 was resolved.

Section 2.2.2.2 of TR-54 refers to Figure 2-2 for the dynamic beam model of a single rack. The text and figure do not adequately describe the model. The staff issued RAI-TR54-24 which reads as follows:

a. Define what each series of nodal degrees-of-freedom (DOFs) correspond to (i.e., nodes 1,2; P1, P2, ...; q4, q5, ..., 1*, 2*, ...). While some of these may be deduced by judgment, the report should clearly define all of these.

b. Explain whether there are 5 nodes and 4 beams along the rack beam model to coincide with the 5 nodes and 4 elements of the fuel assemblies?

In a letter dated April 10, 2007, the applicant provided the following response:

a. Table 54-24.1 defines the nodal DOFs for the dynamic beam model of a single rack as depicted in Figure 2-2 of the technical report.

b. The rack cell structure is modeled as a single beam between two nodes, which are located at the top of the rack and at the baseplate elevation. This is consistent with HOLTEC’s standard model for seismic analysis of spent fuel racks, which has reviewed and approved by the NRC on numerous dockets. Although there is not a one to one correspondence between beam nodes and fuel assembly nodes, fuel to cell wall impact loads, which can occur at Elevation 0, 0.25H, 0.5H, 0.75H, and H (where H is the height of the cell structure), are properly transmitted to the rack beam in accordance with the methodology outlined in TR-54 Reference 11.

The staff found part (a) of the response acceptable, and determined that part (b) would require a review of Reference 11. During the April 2007 audit, the applicant indicated that to satisfy part (a) of the response it will include the information provided in the response in the TR. For part (b,) the applicant provided an explanation of how the fuel-to-cell impact terms at each fuel mass elevation are related to the single beam shape function at the same elevation.

During the October 2007 audit, the applicant indicated that Table 2-18 would be added to TR-54. This table defines the nodal DOFs for a single rack dynamic model. The staff reviewed the proposed revisions to Section 2.2.2.2 and Table 2-18. The staff found this technically acceptable. The staff also reviewed the approach described in Reference 11 of technical report APP-GW-GLR-033, Revision 0, for the use of the coupling terms to relate the internal deformations of the fuel nodes to the external deformations of the fuel rack beam nodes at the two ends. Based on this review and the use of this method in prior licensing submittals to the NRC, the staff found this approach to be acceptable.

The staff determined that the applicant made appropriate changes in TR-54, Revision 1. Therefore, RAI-TR54-24 was resolved.

In RAI-TR54-25, the staff requested the applicant to explain whether only full fuel racks are included in the two simulations, or if several scenarios are considered (i. e., different fill ratios, from partially full to full within a given rack; varying fuel locations within the partially filled rack; varying fill and locations in adjacent racks); and to provide the technical justification if only full racks are considered. The staff also asked whether it would be possible to have less than all fuel racks (8) in the pool. If so, then additional simulations would be needed. If not, is there a requirement in the DCD that specifies all fuel racks must always be in place whenever fuel is stored in any of the racks?

In a letter dated May 17, 2007, the applicant stated that all spent fuel racks, in both simulations, are assumed to be fully loaded with maximum weight fuel assemblies. This scenario bounds any partially loaded configuration since it (1) maximizes the vertical compression and lateral friction loads on the support pedestals and (2) produces the maximum rack displacements and fuel to cell wall impacts. The displacements are larger for a fully loaded rack, as opposed to a partially filled rack, because the dynamic model conservatively assumes that all stored fuel assemblies rattle in unison. Hence, the momentum transferred between the rattling fuel mass and the spent fuel rack is at a maximum for a fully loaded rack. For a partially filled rack, the decrease in rattling fuel mass outstrips the destabilizing effect of an eccentric fuel loading pattern.

The applicant further stated that the SFP rack analysis was performed with all eight fuel racks installed during operation of the SFP, which is consistent with the design intent of the AP1000 Spent Fuel Storage Racks. DCD Section 9.1 will include the statement that all spent fuel racks will be in place in the SFP whenever fuel is stored in the spent fuel racks.

The staff reviewed the response and concluded that the explanation provided in the response appears to support the conclusion that fully loaded racks would be expected to maximize impact forces and displacements. In addition, the use of the maximum weight for the fuel assemblies, the analysis assumption that all stored fuel assemblies rattle in unison, and consideration of the upper and lower bound COF at all support legs provide added conservatism to bound the results.

However, the staff noted that Section 9.1.2.2.1 of DCD Revision 16, states “The purchase specification for the spent fuel storage racks requires the vendor to perform confirmatory dynamic and stress analyses. The seismic and stress analyses of the spent fuel racks consider the various conditions of full, partially filled, and empty fuel assembly loadings.” Therefore, the applicant needed to explain this inconsistency between the DCD and the analyses actually performed. The staff noted that this statement occurs in two locations in DCD Section 9.1.

During the October 2007 audit, the applicant indicated that HOLTEC would perform additional analyses considering partially filled racks (from empty to full condition of various racks) in order to address the requirements in the DCD.

During the May 2008 audit, the staff reviewed HOLTEC Report No. HI-2063523, Revision 2, and draft TR-54, Revision 2, which show that additional cases were analyzed, which considered different racks in the pool having varying fill of fuel assemblies ranging from 0 percent, 25 percent, 33 percent, 66 percent, 75 percent, and 100 percent fill. These results, along with the other variations were presented in the report, and subsequent stress evaluations for the racks were based on the worst-case results from all nine cases considered.

TR-54, Revision 2, and related sections in DCD Revision 17 indicated that additional analysis for mixed loading of spent fuel racks was considered. Table 2-4 in TR-54, Revision 2, was revised to add three more simulations consisting of “fully loaded with modified gaps,” “mixed loadings,” and “empty” rack cases. Based on its review of these additional analyses, the staff requested the applicant to address the following items:

1. The report does not describe these additional cases, as discussed during the May 2008 audit. The staff’s understanding from the audit was that in the mixed loading case: (1) different racks had varying fill from 0 percent through 100 percent; and (2) the partially filled racks had the fill offset (i.e., center of mass was offset from the centerline of the rack in the computer model) to simulate the possibility of placing fuel assemblies closer to the side of the fuel rack rather than uniformly distributed. Provide the following information in TR-54: (1) the fill used for each rack and the location of the assumed fuel assemblies to represent possible offsets in the Run 5 simulation (mixed loading case); (2) the modified gaps used in the Run 4 simulation versus the gaps used for the other cases; and (3) a full description of the three additional analyses (Runs 4, 5, and 9). The description should discuss the purpose, approach, results in comparison to the other cases, and clearly state that any subsequent stress evaluations for the racks were based on the worst-case results from all nine runs.

2. Explain why the tabulated displacements in Table 2-10 have not been revised in view of the statement made in Section 2.8.1.4 of TR-54, Revision 2 that the racks did impact the pool walls. The statement implies that the racks displace at least 8.13 cm (3.2 in), which is the available gap between the racks and the pool walls shown on Figure 2-1. In addition, if the displacement of Rack A1, adjacent to the tool storage area, is greater than 8.13 cm (3.2 in) in the south direction, then the 86.36 cm (34 in) referenced clearance shown on Figure 2-1 would need to be revised to ensure that the rack does not displace into the tool storage area boundary.

During the August 2009 audit, the staff discussed with the applicant the need to address the items described above. In a letter dated September 14, 2009, the applicant submitted a revision to its RAI response, addressing these items. The staff confirmed that the necessary changes to TR-54 are included in TR-54, Revision 3. Therefore, RAI-TR54-25 was resolved.

In RAI-TR54-26, the staff requested the applicant address the following: what are the gaps and tolerances for each of the gaps between the fuel to cell wall, rack to rack, and rack to wall? What are the assumed initial locations of the various components (fuel assemblies and each rack) and what is the technical basis for this assumption? Were any studies done for different initial conditions (considering tolerances); if not, explain why. What requirements are in the DCD to ensure that the assumed gaps (considering tolerances) will always be maintained throughout the licensing period?

In a letter dated May 17, 2007, the applicant stated that all gaps between fuel assemblies and cell walls, between racks, and between racks and pool walls are set to match the nominal gaps provided on the Westinghouse Drawing APP-FS02-V2-002, Revision 0, “Discrete Zone Two Region Spent Fuel Rack Pool Layout.” An attached table summarized the gap information used in the dynamic analyses. The applicant further stated that fuel is assumed centrally located in the cell. This is conservative since minimizing gap on one or two walls will generally produce a larger hydrodynamic coupling effect. Numerical studies were done on other HOLTEC rack projects; the studies generally showed a small influence on results. A larger influence occurs if the gaps are assumed to be displacement-dependent, rather than always being held constant at their initial value. Neglecting this effect is conservative.

The applicant also stated that, once racks are installed in the SFP, the only way the rack-to-rack and rack-to-wall gaps would change over time would be by the action of a seismic event. COL applicants will have a procedure in place to measure the post-earthquake gaps, to evaluate the acceptability of the configuration, or to take appropriate corrective actions.

During the May 2008 audit, the staff reviewed HOLTEC Report HI-2063523, Revision 2, and confirmed that an additional case (Run 4) was analyzed to consider the effect of the installation tolerances for the nominal gaps. Run 4 corresponded to the base case conditions (Run 1 - COF equal to 0.8, fully loaded racks, integration time step = 1×10^{-5}) but it increased all of the gaps between adjacent racks by 1.27 cm (0.5 in). The results of this additional run demonstrated that the response of the racks was within 8 percent of the base case except for the maximum shear force on the rack pedestal, which was 32 percent higher. The applicant demonstrated that even with this higher pedestal shear force, the pedestal stress factor (ratio of calculated maximum shear stress to allowable shear stress) is equal to 0.092, which is very low. For completeness, the results of Run 4 were included in the design of the racks.

During the August 2009 audit, the staff determined that the drawings showing gaps and tolerances needed to be revised again, to reflect the latest analysis results for revised seismic loading. Subsequent rack design changes and a final revision to the seismic loading, following the August 2009 audit, required further revision of the drawings.

During the June 2010 audit, the staff and the applicant reviewed the final gaps and tolerances. In a letter dated July 20, 2010, the applicant submitted a revised RAI response that addressed the applicable figures in TR-54, Revision 4. The staff reviewed the response, and found the figures to be consistent with the gap and tolerance information in TR-54, Revision 4 and presented at the June 2010 audit.

In RAI-TR54-27, the staff requested the applicant provide more detailed information about how the fluid coupling was calculated and implemented in the AP1000 simulations. Describe the approaches used for fluid coupling of fuel assemblies to fuel cell walls, rack to rack, and rack to pool wall because there would be some differences among these. For the rack to rack and rack to wall fluid coupling, explain how fluid flow was considered horizontally as well as vertically over the top of the racks and flow to the bottom of the rack. Describe and justify any assumptions made in the approach. For example, small vibratory deflections relative to the gaps are probably assumed and the fluid gaps are not updated according to the rack displacements (see Section 2.4 of this report).

In a letter dated May 17, 2007, the applicant provided the following response:

> A mathematical explanation of the manner in which fluid coupling is calculated and implemented in the AP1000 simulations is provided.
>
> The problem to be investigated is shown in Figure TR54-27.1, which shows an orthogonal array of 8 rectangles which represent a unit depth of the 8 spent fuel racks in the AP1000 Spent Fuel Pool. The rectangles are surrounded by narrow fluid filled channels whose width is much smaller than the characteristic length or width of any of the racks. The spent fuel pool walls are shown enclosing the entire array of racks.
>
> The dimensions of the channels are such that an assumption of uni-directional fluid flow in a channel is an engineering assumption consistent with classical fluid mechanics principles. Each rectangular body (fuel rack) has horizontal velocity components U and V parallel to the x and y axes, and the channels are parallel to either the x or y axes. The pool walls are also assumed to move.
>
> It is conservatively assumed that the channels are filled with an inviscid, incompressible fluid. Due to a seismic event, the pool walls and the spent fuel racks are subject to inertia forces that induce motion to the rectangular racks and to the walls. This motion causes the channel widths to depart from their initial nominal values and causes flow to occur in each of the channels. Because all of the channels are connected, the equations of classical fluid mechanics can be used to establish the fluid velocity (hence, the fluid kinetic energy) in terms of the motion of the spent fuel racks.
>
> For the case in question, there are 22 channels of fluid identified.
> Figure TR54-27.2 shows a typical rack (box) with four adjacent boxes with the fluid and box velocities identified. [See Westinghouse response for calculation and figures.]
>
> There are a total of 15 + 8 = 23 equations which can be formally written; one circulation equation, however, is not independent of the other. This reflects the fact that the sum total of the 8 circulation equations must also equal zero, representing the fact that the circulation around a path enclosing all racks is equal to zero. Thus, there are exactly 22 independent algebraic equations to determine the 22 unknown mean velocities in this configuration.
>
> Once the velocities are determined in terms of the rack motion, the kinetic energy can be written and the fluid mass matrix identified using the HOLTEC

> International QA validated pre processor program CHANBP6. The fluid mass matrix is subsequently apportioned between the upper and lower portions of the actual rack in a manner consistent with the assumed rack deformation shape as a function of height in each of the two horizontal directions. The HOLTEC International pre processor program VMCHANGE performs this operation. Finally, structural mass effects and the hydrodynamic effect from fluid within the narrow annulus in each cell between the fuel assembly and the cell wall are incorporated using the HOLTEC International pre processor program MULTI155.

The staff reviewed the RAI response and concluded that the response only addressed how fluid coupling of rack to rack and rack to wall is calculated and implemented in the AP1000 simulation. As requested in the RAI, the applicant should also provide a description of the approach used for simulating the fluid coupling of the fuel assemblies to the fuel cell walls. Due to the complex nature of the fluid coupling approach used for the rack to rack and rack to wall, and the use of several HOLTEC in-house computer codes, how has the approach been verified (e.g., test data or alternate analytical methods with known solutions)?

In a letter dated April 18, 2008, the applicant revised its response, replacing the last two paragraphs, and adding references as shown below:

> "Once the velocities are determined in terms of the rack motion, the kinetic energy can be written and the fluid mass matrix identified using the HOLTEC International QA validated pre processor program CHANBP6. The fluid mass matrix is subsequently apportioned between the upper and lower portions of the actual rack in a manner consistent with the assumed rack deformation shape as a function of height in each of the two horizontal directions. The HOLTEC International pre processor program VMCHANGE performs this operation.
>
> "The approach used for fluid coupling between the fuel assemblies and the cell walls is presented in Reference 2, and it is based upon Fritz's classical two body fluid coupling model (Reference 3). References 2 and 3 were previously provided to the NRC as part of the April 9, 2007 RAI response submittal (Reference 4 - Westinghouse Letter DCP/NRC1860). The structural mass effects and the hydrodynamic effect from fluid within the narrow annulus in each cell between the fuel assembly and the cell wall is incorporated using the HOLTEC International preprocessor program MULTI155."

The staff noted that the fluid coupling between the fuel assemblies and the cell walls is described in a paper "Seismic Responses of Free Standing Fuel Rack Constructions to 3 D Motions," by Soler, A.I. and Singh, K.P., Nuclear Engineering and Design, Volume 80, pp. 315 329 (1984), and it is based upon Fritz's classical two-body fluid coupling model presented in "The Effects of Liquids on the Dynamic Motions of Immersed Solids," by Fritz, R.J., Journal of Engineering for Industry, Trans. of the ASME, February 1972, pp. 167-172. This methodology for modeling the fluid coupling effects has been accepted for predicting the response of spent fuel racks in NUREG/CR-5912. It has been used in past licensing of spent fuel racking, and reviewed and approved by the NRC. Therefore, RAI-TR54-27 was resolved.

The load combinations specified in Table 2-5 of TR-54 and Table 9.1-1 (markup version of the DCD provided with the subject report) did not match NUREG-0800 Section 3.8.4, Appendix D criteria. The staff issued RAI-TR54-29, which reads as follows:

a. No load combinations are specified for the spent fuel racks corresponding to service Level A.

b. Temperature conditions T_o and T_a are not included in Table 2-5; however, they are included in the markup DCD Table 9.1-1. A footnote in the markup of DCD Table 9.1-1 states that "For the faulted load combination, thermal loads will be neglected when they are secondary and self limiting in nature and the material is ductile. In freestanding spent fuel racks, thermal effects mainly affect the temperature that is used in specifying the allowable stress and Young's Modulus." Based on this statement:

   (i) Regarding the first quoted sentence above, Table 2-5, Load Combination corresponding to service levels A and B (which are not the faulted condition) should include T_o.

   (ii) Regarding the last quoted sentence above, SRP 3.8.4, Appendix D indicates that thermal loads due to temperature effects and temperature gradients across the rack structure need to be considered. Temperature gradients can occur due to differential heating effects between one or more filled cell(s) and one or more adjacent empty cell(s). The stresses from these types of thermal loads should be considered because they can still lead to localized failure of the structure. When responding to this, consider temperature loads due to normal and accident conditions, as noted in your Table 9.1-1 and SRP 3.8.4, Appendix D.

c. Table 2-5 in the report and DCD Table 9.1-1 indicate that the load term P_f is the uplift force on the rack caused by a postulated stuck fuel assembly accident condition or the force developed on the rack from the drop of a fuel assembly during handling to the top of the rack or the baseplate through an empty cell. SRP 3.8.4, Appendix D, separates these two accident events into Pf for the uplift force event and Pd for the drop load event. This is necessary because SRP 3.8.4, Appendix D specifies that the acceptance limits for these two events (in combination with deadweight + live load + thermal) are different.

d. Table 2-5, last load combination with E', does not provide the Service Limit. If the same Service Limit, D(1), as indicated in the load combination above the last load combination was intended, then explain whether the functionality capability requirement in footnote (1) (which is applicable to only the new racks) is in addition to or in-place of Level D limits.

In a letter dated May 17, 2007, the applicant provided the following response:

> Table 2-5 of technical report (TR)-54 and DCD Table 9.1 1 will be revised as follows (which is derived from Appendix D to SRP Section 3.8.4):
>
> a. Table 2 5 of the subject report and DCD Table 9.1 1 will be modified to specify the load combinations D + L and D + L + T_o for Service Level A, as shown above.

b. (i) Table 2 5 of the subject report will be modified to include T_o for Service Levels A and B, as shown above.

   (ii) The temperature gradients across the rack structure caused by differential heating effects between one or more filled cells and one or more adjacent empty cells are considered. The worst thermal stress field in a fuel rack is obtained when an isolated storage location has a fuel assembly generating heat at maximum postulated rate and the surrounding storage locations contain no fuel. This secondary stress condition is evaluated alone and not combined with primary stresses from other load conditions. A thermal gradient between cells will develop when an isolated storage location contains a fuel assembly emitting maximum postulated heat, while the surrounding locations are empty. A conservative estimate of the weld stresses along the length of an isolated hot cell is obtained by considering a beam strip uniformly heated by 10.00 °C (50 °F), and restrained from growth along one long edge. The 50 °F temperature rise envelops the difference between the maximum local spent fuel pool water temperature (174 °F) inside a storage cell and the bulk pool temperature (140 °F) based on the thermal hydraulic analysis of the spent fuel pool. The cell wall configuration considered here is shown in figure below.

c. The definition of P_f in Table 2-5 of the subject report and DCD Table 9.1-1 is incorrect. The referenced tables will be revised to clearly distinguish between P_f and F_d.

d. Level D service limits apply to load combination D + L + Ta + E'. Per Appendix D of SRP Section 3.8.4, the functional capability of the fuel racks should be demonstrated for the accidental drop event (D + L + F_d). This requirement is in place of the Level D service limits since it is recognized that the rack may sustain permanent damage due to the impact force, and therefore it may not be possible to meet Level D service limits at all locations within the rack. The functional capability of the spent fuel racks is generally defined as the continued ability of rack to store spent fuel assemblies in a subcritical arrangement.

The staff reviewed the RAI response, and concluded that several additional items needed to be addressed by the applicant. In a letter dated April 18, 2008, the applicant provided a revised response to this RAI, which provided additional explanations.

The staff reviewed the revised response, and noted that proposed TR-54 Table 2-5 and DCD Table 9.1-1 include the missing load combination. In addition, the applicant: (1) explained why a secondary stress is not combined with primary stresses from other loads; (2) provided the calculation for the analysis of thermal stresses in the rack cell considering the differential heating between a cell containing a fuel assembly emitting maximum postulated heat and empty cells surrounding the heated cell; and (3) explained that the shear stress in the weld is caused by thermal loading, which is classified by the ASME Code as a secondary stress. In summary, the staff accepted the applicant's explanations pending submittal of revisions to TR-54 and the DCD. The staff subsequently confirmed that TR-54, Revision 2, and DCD Revision 17 had been appropriately revised. Therefore, RAI-TR54-29 was resolved.

The staff noted that TR-54 does not address seismic-induced sloshing effects. In RAI-TR54-31, the staff requested that the applicant provide a description of the sloshing calculation approach and results for both horizontal directions.

In a letter date May 17, 2007, the applicant stated that “sloshing” may be defined as the dynamic behavior and associated load of the water produced by wave-like motion at the surface of the pool. TID-7024, “Nuclear Reactors and Earthquakes,” Chapter 6, is commonly used to evaluate the dynamic response of the water within the SFP. Figure 6.2(a) of TID-7024 depicts the two masses of water that the total bulk is considered to be split into, as described in the text. The upper portion of the water, denoted in the figure as “water in motion,” produces convective forces and the lower portion of the water, denoted as “constrained water,” produces impulsive forces. The latter bulk of water has an associated mass (identified as weight W0) and is effectively a rigid body that moves along with the tank (refer to Figure 6.1 and the first paragraph of Section 6.4). The horizontal force produced by this mass of water when accelerated by the earthquake acts at a height of h_o from the bottom of the tank. This parameter is determined in the table given at the end of Section 6.3 to be equal to 3/8 times the height of the fluid. This height is not dependent upon the magnitude of the earthquake. For the SFP, the water depth is approximately 12.19 m (40 ft) and the height h_o would be 4.57 m (15 ft) from the bottom. Since the impulsive force acts at the approximate centroid of the rigid water mass, the top elevation of this bulk of water is above this point. As the racks are approximately 515.6 cm (203 in) tall, which is only slightly higher than the height h_o, the racks reside in the impulsive water mass at the bottom of the pool and the sloshing portion of the water is above this elevation. Therefore, seismic sloshing of the SFP water does not influence the dynamic response of the spent fuel racks in either horizontal direction.

The staff concluded that the applicant’s description of the sloshing effect in the SFP, based on using the method presented in TID-7024, Chapter 6, demonstrates that the racks reside in the impulsive water mass region of the pool and the sloshing portion of the water is above the top of the racks. Therefore, RAI-TR54-31 was resolved.

Section 2.3.4.3 of TR-54, fourth bullet, develops the faulted allowable maximum weld stress for the weld material. In RAI-TR54-33, the staff asked the applicant why an allowable maximum weld stress based on the base metal isn’t also developed. Normally welds are checked for both weld material and base metal, as was done for Levels A and B in Section 2.3.4.1.

In a letter dated May 17, 2007, the applicant stated:

> The required capacity evaluation for Level A conditions are presented below using the material properties associated with the material. Using the ASME Code allowable strengths for weld and base metal in Subsection NF, the shear capacities are:
>
> V(base) = (0.4Sy)Al; V(throat) = (0.3Su)(0.707Al);
> V(throat)/V(base) = 0.2121Su/(0.4Sy) = 0.53025Su/Sy, where Su = ultimate strength of weld material (assumed equal to that of the base metal for purposes of this calculation); Sy = yield strength of base metal; Al = fillet weld leg area; At = fillet weld throat area = 0.707Al.
>
> The above result for Level A conditions shows that the weld throat controls the capacity only if 0.53025Su < Sy. For the AP1000 spent fuel racks, Su=66.2 ksi; Sy=21.3 ksi at temperature, so that V(throat)/V(base) = 1.648, indicating that

base shear capacity controls the joint for a Level A event. For Levels B, C, and D, the joint capacities are simply increased by a factor so that the determination of the governing section remains the same.

Appendix F of the ASME Code does not explicitly require weld calculations for Level D events. If, however, the weld capacity evaluations are performed using material strengths inferred by certain sub sections of Appendix F, HOLTEC evaluates the capacity of the weld throat by using the amplifier 1.8 on the Level A capacity to obtain V(throat) = 1.8 (0.2121SuAl) = 0.38278SuAl

ASME Code Appendix F contains the following subsections that refer to allowable strengths for shear calculations. Using the 1998 Edition,
F-1331 - Criteria for Components (F-1331.1(d)) - The average primary shear stress across a section loaded in pure shear shall not exceed 0.42Su.
F-1332 - Criteria for Plate and Shell Type Supports (F-1332.4 Pure Shear) - The average primary shear stress across a section loaded in pure shear shall not exceed 0.42Su.
F-1334 - Criteria for Linear Type Supports (F-1334.2 Stresses in Shear) - The shear stress on the gross section shall not exceed the lesser of 0.72Sy and 0.42Su. Gross section shall be determined in accordance with NF 3322.1(b). [Note that Code reference to NB 3322.1(b) is a typo as the referenced NB section has nothing to do with section evaluation.]

It is stipulated that F-1334.2 is intended for setting limits for the shear stress in the base metal of gross sections associated with steel structural members and should not be applied to any weld calculation (as can be inferred by the title of Subsection NF-3322 B Design Requirements for Structural Steel Members). Even if one accepts that there is an implied requirement in Appendix F to check weld capacity for Level D events, the appropriate base metal shear stress limit should be 0.42Su (viz. F-1331.1(d), F-1332.4, or F-1334.2), which would therefore give the capacity of the base metal as V(base) = 0.42SuAl.

V(throat)/V(base) = 0.911 indicating that weld throat shear capacity always controls the joint for a Level D event independent of the material. This is why only the weld throat is checked when examining welds in the Level D configuration.

The staff reviewed the above explanation and found that it demonstrates that following the requirements in Appendix F for components and supports, and using the ultimate strength value and yield strengths for the materials of the spent fuel racks, welded joints are governed by the weld throat shear capacity, not the base metal capacity. That is why only the weld throat was checked for Level D when examining the structural adequacy of welds. The staff found this acceptable. Therefore, RAI-TR54-33 was resolved.

Section 2.3.5 of TR-54 discusses dimensionless stress factors. It states that “R1 is the ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression).” In RAI-TR54-34, the staff requested the applicant explain why this indicates that pedestals only resist compression, since horizontal forces are also generated due to friction during a seismic event. These forces could be quite high and also would introduce shear and moments into the pedestal and rack structure.

In its response, the applicant stated that Section 2.3.5 of the report defines seven stress factors (R1 through R7), which correspond to the ASME Code Section III, Subsection NF stress limits for Class 3 components. R1 is defined as the ratio of direct tensile or compressive stress on a net section to its allowable value. Since the spent fuel racks are freestanding, the net cross section of the support pedestals can only be subjected to direct compressive stress. This is the explanation for the note in parentheses. The applicant further stated that horizontal forces are generated due to friction between the support pedestals and the SFP floor and that these forces produce shear and bending stresses in the pedestals. The shear and bending stresses in the support pedestals, as well as the combined compression and bending stress, are measured by the other six stress factors (i.e., R2 through R7), which are defined in Section 2.3.5 of this report. The staff reviewed the RAI response and concluded that the explanation is acceptable. Therefore, RAI-TR54-34 was resolved.

Section 2.8.1.4 of TR-54 describes the impact loads and states that these loads do not result in damage to the racks that would prevent retrievability. In RAI-TR54-35, the staff requested the applicant confirm that the acceptance criteria for these impacts include both retrievability and meeting the stress limits for Level D in accordance with ASME Code, Section III, Subsection NF; and to provide the stress ratios for the most critical cells adjacent to the worst case impact.

In a letter dated May 17, 2007, the applicant stated that the ability to retrieve the fuel is based solely on evaluating the rack structure to show that there is no instability that would collapse the cell. Subsection NF stress limits for Level D do not apply to the local stress state in the impacted cells because: (a) the fuel racks are analyzed as linear type supports (i.e., beam type members) in accordance with Appendix D of NUREG-0800 Section 3.8.4; and (b) rack to rack impact loads near the top of the rack produce secondary stresses, for which there is no prescribed limit in ASME Code, Section III, Subsection NF for Level D. Away from the point of impact, the rack to rack impact loads do produce primary bending and shear stresses in the rack beam, which are reflected in the maximum stress factors reported in TR-54, Table 2-9.

The staff reviewed the response, and requested additional information explaining what acceptance criteria are used to ensure that the fuel assemblies can be retrieved following the impact: If a quantitative stress limit criteria is not utilized, then what is the specific criterion used (e.g., are the cross-sectional opening dimensions in each cell checked for any permanent deformation that would infringe on the fuel assembly outside dimension)?

In a letter dated April 18, 2008, the applicant stated that in order to ensure fuel retrievability is maintained, the impact loads at the rack top elevation are compared against 2/3 of the critical buckling load for the cell walls, as required by Table NF-3523(b)-1 of the ASME Code for primary plus secondary stresses. The critical buckling load calculation is included in HOLTEC Report No. HI-2063523, “Spent Fuel Rack Structural/Seismic Analysis for Westinghouse AP1000,” Revision 0. The impact load is assumed to spread uniformly over a 15.24 cm (6 in) vertical span of the cell wall, which is equal to the minimum length of the intermittent cell to cell welds. The average compressive stress in the cell walls due to the maximum rack-to-rack impact load is 132 MPa (19,120 psi). This stress is less than two thirds of the critical buckling load. Therefore, the spent fuel rack design meets the requirements of Table NF-3523(b)-1 of the ASME Code for the primary plus secondary stress category.

The staff concluded that the calculation adequately demonstrated that the stress due to the maximum impact load is below 2/3 of the critical buckling load, and that the spent fuel racks meet the requirements of Table NF-3523(b)-1 for primary plus secondary stress category, which covers buckling. The staff found this approach to demonstrate retrievability acceptable. The staff requested the applicant expand the limited one-sentence conclusion presented in Section 2.8.1.4 of the TR-54, to summarize the information in the RAI response and the buckling calculation. The staff subsequently reviewed TR-54, Revision 2, and related sections in DCD Revision 17, to confirm the inclusion of additional descriptive information. Therefore, RAI-TR54-35 was resolved.

Some of the information provided in Section 2.8.2 (Rack Structural Evaluation) and Tables 2-9 through 2-15 (stress results) of TR-54, Revision 0, was not clear. The staff issued RAI-TR54-36 which reads as follows:

a. Section 2.8.2.1, 2nd paragraph, indicates that the tables also report the stress factors for the AP1000 Spent Fuel Storage Racks cellular cross section just above and below the baseplate. This implies that the fuel cells continue below the baseplate. Explain.

b. The same paragraph refers to "pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)." Explain what this means since the tables do not reflect this terminology.

c. The same paragraph refers to "ensures that the overall structural criteria set forth in Subsection 2.2.3 are met." Structural criteria are not presented in Subsection 2.2.3.

d. Section 2.8.2.2 a., refers to a stress factor of 2.1516 which it states is given in the tables. However, no such stress factor is given, please explain. Also, are all cells welded to the baseplate on all four sides?

e. Section 2.8.2.2 a., first bullet, calculates the stress in the weld, connecting the cell walls to the baseplate, equal to 25,047 psi; however, Table 2-12 shows a smaller (maximum) weld stress of 22,647. Explain.

f. Section 2.8.2.2 b. indicates that a separate finite element model is used to check the baseplate to pedestal welds. Provide a short description of the model, computer code, loading, and location of the maximum tabulated stress in the weld referred to in Table 2-14.

g. Section 2.8.2.2 c. indicates that for calculation of cell welds, the fuel assemblies in adjacent cells are conservatively calculated by assuming that the fuel assemblies in adjacent cells are moving out of phase with one another. It then states that cell to cell weld calculations are based on the maximum stress factor from all runs. However, elsewhere in the report it was stated that all of the fuel assemblies in the simulation are assumed to vibrate in phase. Provide more information to explain this. Also, this paragraph indicates that both the weld and the base metal shear results (for cell to cell) are reported in Table 2-14; however, Table 2-14 is labeled baseplate to pedestal welds. If reference was intended to Table 2-15, then note that Table 2-15 provides the shear stress only for the base metal.

h. Section 2.8.2.3 refers to Tables 2-9 through 2-14 for limiting pedestal thread shear stresses for every pedestal. These tables do not seem to apply to pedestal thread shear stress. Therefore, clarify or correct this information.

i. Table 2-9, Summary: identify what rack component/element applies to each of the column headings (i.e., Max. Stress Factor, Max. Shear Load, Max. Fuel to Cell Wall Impact). Similarly, for the other tables, identify what rack component/element the table applies to (e.g., Tables 2-13 and 2-15 are missing this information).

j. Table 2-10 provides maximum rack-to-rack displacements relative to the floor. Also provide maximum & minimum relative displacements to the walls.

k. Why are results for "Run 1 and 2" given for some tables and not others? Both should be provided or an explanation should be given why they are included for some tables and not for others.

l. Table 2-15, why is this table labeled “Allowable Shear Stress ...” versus the labeling of other tables and why is it labeled Level D, versus other tables where there is no indication of Levels? All tables should identify which load level they apply to.

In a letter dated June 14, 2007, the applicant provided the following response:

a. The fuel cells do not continue below the baseplate. Stress factors are computed just above the baseplate, where the fuel cells are welded to the baseplate and just below the baseplate where the support pedestals are welded. Section 2.8.2.1 (2nd paragraph, 2nd sentence) will be revised as follows:

   The tables also report the stress factors for the AP1000 Spent Fuel Storage Racks cellular cross section just above the baseplate.

b. The computer code DYNAPOST, which is listed in Table 2-8, computes the stress factors for the four support pedestals and for the cellular structure just above the baseplate based on the time history analysis results. For convenience, these five locations are identified as pedestal numbers 1 through 5 in the DYNAPOST output tables, which are not included in technical report APP-GW-GLR-033. Therefore, the sentence, "The locations above the base plate ... are referred to as pedestal five in the first sheet of the summary tables for each simulation (that is, 9.M.0 where M stands for run number)." is not relevant to the report and will be deleted.

c. The reference to Subsection 2.2.3 is a typo. The correct reference is Subsection 2.3.3.

d. The factor of 2.1516 is not provided in the tables as stated in text. Section 2.8.2.2 a. (2nd paragraph) will be revised as follows:

   Weld stresses are determined through the use of a simple conversion (ratio) factor (based on area ratios) applied to the corresponding stress factor in the adjacent rack material. This conversion factor is developed from the differences in base material thickness and length versus weld throat dimension and length:

   All fuel cells are welded to the baseplate on all four sides.

e. The correct stress in the weld is 25,047 psi. Table 2 12 will be revised to change 22,647 psi to 25,047 psi, as shown [in Table 2-12 of the RAI response].

f. The finite element code ANSYS is used to resolve the tension and compression stresses in the pedestal weld due to the combined effects of a vertical compressive load in the pedestal and a bending moment caused by pedestal friction. The compression interface between the baseplate and the pedestal is modeled using contact elements. The perimeter nodes on the pedestal are connected to the baseplate by spring elements in order to simulate tension in the weld. The maximum instantaneous friction force on a single pedestal from the rack seismic analysis is conservatively applied to the finite element model in the horizontal x and y directions simultaneously, along with the concurrent vertical load, at the appropriate offset location. The perimeter nodes on the pedestal are restrained to move only in the vertical direction so that the spring elements only resist bending. The limiting ANSYS results are combined with the maximum horizontal shear loads to obtain the maximum weld stress. The maximum weld stress reported in Table 2-14 occurs at the corner of the pedestal where the tensile stress in the weld due to bending is at its maximum.

g. All stored fuel assemblies within a rack are assumed to rattle in phase for the seismic analysis of the spent fuel racks using the HOLTEC proprietary computer code MR216 (a.k.a. DYNARACK). This analysis yields the maximum impact force between a single fuel assembly and the surrounding cell walls. When evaluating the weld connection between adjacent storage cells, the maximum fuel to cell impact force from the dynamic analysis is conservatively multiplied by a factor of 2 to consider out of phase fuel rattling. The reference to Table 2-14 in Section 2.8.2.2 c is incorrect. The shear stress results for the cell to cell weld connection are not provided in Table 2-14 or Table 2-15. The shear stress in the cell to cell weld and the adjacent base metal are 11,646 psi and 8,235 psi, respectively. The allowable stress limits are 35,748 psi and 18,000 psi, respectively. Tables 2-16 and 2-17 (see below) will be added to technical report APP-GW-GLR-033 to provide the shear stress results for the cell to cell weld and the adjacent base metal, respectively.

h. The reference to "Tables 2-9 through 2-14" in Section 2.8.2.3 is incorrect. The first sentence in Section 2.8.2.3 will be revised as follows: "Table 2-15 provides the limiting thread stress under faulted conditions."

i. In Table 2-9, the "Max. Stress Factor" column applies to the rack cell structure. The "Max. Vertical Load" and "Max. Shear Load" columns apply to a single rack pedestal. The "Max. Fuel to Cell Wall Impact" column provides the maximum impact force between a single fuel assembly and the surrounding cell wall at any of the five rattling fuel mass elevations (refer to Figure 2-5 of the report).

   Table 2-13 applies to the base metal adjacent to the baseplate to cell welds. Table 2-15 applies to the pedestal internal threads.

j. Table 2-10 provides the maximum displacement in any direction (x or y) for all racks, relative to the floor. In other words, the rack displacements in Table 2-10 are the bounding displacements for all rack to rack and rack to wall gaps. The results in Table 2-10 also represent the maximum rack displacements relative to the pool walls since the SFP structure is assumed to be rigid for the purpose of the rack seismic analysis (i.e., the SFP floor and walls displace equally). The minimum rack displacement relative to the SFP walls (which is interpreted as maximum distance that a rack displaces away from the SFP walls) is also bounded by the results in Table 2-10, since the reported displacements are the maximum (absolute value) displacements for all racks.

k. The stress results in Tables 2-12 through 2-15 are the maximum values from Run 1 and Run 2.

l. Table 2-15 should be labeled "Pedestal Thread Shear Stress" instead of "Allowable Shear Stress for Level D". The allowable stresses reported in Tables 2-12 through 2-15 are Level D stress limits since the design basis ASB99 earthquake is a faulted condition (Level D).

The staff reviewed the response, and concluded that for parts (a) through (d), (h), (i), (k), and (l), the clarifications and editorial corrections are acceptable; however, revision of TR-54 would be required as noted in the response. For part (e), correction of the stress result in Table 2-12 is acceptable; however, with this correction the safety factor noted in the table is no longer correct and needs to be revised. For part (f), the description of the separate finite element model to check the baseplate to pedestal welds is acceptable and should be included in the next revision of TR-54. For part (g), the response explained why in the dynamic analyses the in-phase assumption for fuel assembly motion was utilized, while for the design of the welds between adjacent cells, the out-of-phase motion of fuel rattling was used. This approach is considered to be acceptable because it would maximize the rack motion and impact forces in the dynamic analyses while the out of phase motion of the fuel assembly would be more conservative for the evaluation of the welds between adjacent cells. However, the shear stress results refer to the wrong table, as indicated in the response, and two new tables have been developed which need to be inserted in TR-54. For part (j), additional clarification is needed to explain: (1) whether the individual maximum displacements are at the base or at any elevation of the rack model including the top of the rack since the rotation of the racks about a leg would amplify the horizontal motion from the base; and (2) how do these displacements compare to the initial available gap to the pool walls and rack-to-rack gaps at the top and bottom. This would demonstrate whether impacts with the pool wall occur.

During the May 2008 audit, the staff reviewed TR-54, Revision 1, and concluded that the applicant’s responses to all parts of this RAI are technically acceptable, and appropriately addressed in the TR and the DCD. Therefore, RAI-TR54-36 was resolved.

Section 2.8.4 of TR-54 indicates that this section presents evaluations for potential cell wall buckling and the secondary stresses produced by temperature effects. The staff noted that the description of secondary stresses produced by temperature effects is not included in this section. In RAI-TR54-37, the staff requested the applicant add this information to the report, and to confirm that the R5 stress factor used for the buckling calculation includes the worst impact forces generated, including the impacts at the top of the racks.

In a letter dated May 17, 2007, the applicant stated that the secondary stresses produced by temperature effects (an isolated hot cell) were inadvertently omitted, and that TR-54 Section 2.8.4 would be revised to include an evaluation of secondary stresses produced by temperature effects. The stress factor R5 is a stress factor that is used to get the vertical stress near the base of a corner cell and includes the effect of lateral impact forces at the top of the rack. That is, at any instant the rack is under beam action so that a lateral impact load at the top of a rack develops a vertical load at the base of the rack as the rack resists rocking.

During the May 2008 audit, the staff reviewed TR-54, Revision 2, and confirmed that the evaluation of the secondary stresses produced by temperature effects was included in Section 2.8.4.2 of the TR. The thermal analysis did not consider the contribution from seismic loads. The applicant explained that this was done because the extreme thermal analysis was based on the conservative case of a single cell with a fuel assembly surrounded by all empty cells. The seismic stress contribution for this rack configuration would be insignificant. The applicant demonstrated this by presenting the results for Run 9, which considered all racks empty. These results showed that the maximum stress factor was 0.074, which is extremely small compared to 1.0. Therefore, RAI-TR54-37 was resolved.

The staff noted that the computer code MR216 (a.k.a. DYNARACK) as well as the other computer analysis codes should have complete validation documentation, available for review during an audit. In RAI-TR54-39, the staff inquired if any of the computer codes have been previously reviewed and approved by the staff on other licensing applications, for the same version of the code.

In its response dated April 9, 2007, the applicant stated that all computer analysis codes used to perform the seismic analysis of the spent fuel racks have been validated in accordance with HOLTEC's 10 CFR 50 Appendix B quality assurance program. The validation documentation will be available for review during the audit. The validation documentation for the computer code MR216 has been previously submitted by HOLTEC International to the NRC staff for review and approval several times. Most recently it was reviewed by the NRC in 1998 in Docket Number 50-382 for the Waterford 3 Steam Electric Station.

During the April 2007 audit, the applicant indicated that the version of the MR216 code previously used on the Waterford 3 Steam Electric Station and the version used for the AP1000 are identical, except that the code was revised to accept an additional input at the top of the structure being analyzed. This change has been validated; however, this feature is not used in the AP1000 analyses.

During the October 2007 audit, the staff reviewed the DYNARACK computer validation package. The validation package provided for staff review was the HOLTEC I.D. No. HI-91700 (Generic), "DYNARACK Validation Manual," Revision 1, approved January 28, 1998. The approach used to validate DYNARACK was to demonstrate that it meets the validation requirements of NUREG-0800 Section 3.8.1. The procedure followed for the validation of the code was Section II.4.F (in the current March 2007 version) of NUREG-0800 Section 3.8.1. A series of validation problems were performed and described in the validation manual, demonstrating that criteria (ii) and (iii) in Section II.4.F were met. The staff reviewed a representative test problem included in the validation package. Based on this review, and the accepted use of DYNRACK on a number of other rack analyses for nuclear power plant licensing submittals, the staff concluded that this validation package for DYNARACK is acceptable.

The staff also reviewed the computer program validation package for the DYNAPOST code. The validation package is HOLTEC Report No. HI-971648, "QA Validation of Program DYNAPOST for Generic," Revision 1, approved October 31, 1997. This program was developed to post process the results obtained from the whole pool multi-rack analysis performed with DYNARACK. Based on the review, the staff concluded that the validation package for DYNAPOST is acceptable.

The staff concluded that the computer codes used for the seismic response analysis of the fuel racks have been validated and supporting documentation exists. Therefore, RAI-TR54-39 was resolved.

In RAI-TR54-40, the staff requested the applicant explain what provisions are provided for performance of inservice examination of the rack, as specified in 10 CFR 50.55a(g)(3) for ASME Code Class 3 component supports.

In a letter dated May 17, 2007, the applicant stated that the spent fuel racks are passive structures in the SFP. They operate in a relatively mild environment compared to reactor coolant system (RCS) primary components. There are no moving parts on the spent fuel racks, and they do not require any instrumentation. Therefore, there is no compelling need to perform inservice examination of the spent fuel racks. However, the spent fuel racks can be accessed from above by way of an empty storage cell location(s) to enable the performance of inservice examination, as mandated by 10 CFR 50.55a(g)(3) for ASME Code Class 3 component supports. At the base of each storage cell (except at the four designated lifting locations), there is a 15.24 cm (6 in) diameter thru-hole in the baseplate, which provides access below the baseplate. Also, access below the baseplate can be gained from the area of the SFP that does not contain spent fuel racks. In summary, the spent fuel racks are designed to provide access to all surfaces that may come in contact with spent fuel assemblies and to the support pedestals beneath the baseplate, to support inservice examinations as needed.

The staff concluded that adequate accessibility has been provided to accommodate inservice inspection of the spent fuel racks. Therefore, RAI-TR54-40 was resolved.

Section 2.1.1 was revised in TR-54, Revision 2, to state that "Per DCD Subsection 3.7.5.2, COL applicants will prepare site-specific procedures for activities following an earthquake. An activity will be to address measurement of the post-seismic event gaps between spent fuel racks and to take appropriate corrective actions." This statement was previously in Section 2.9 "Conclusions," in TR-54, Revision 0 and Revision 1, and was moved to Section 2.1.1 in TR-54, Revision 2.

The staff noted that DCD Section 3.7.5.2 does not discuss the need for COL applicants to prepare site-specific procedures for checking the gaps between the fuel racks following an earthquake. In RAI-SRP9.1.2-SEB1-04, the staff requested the applicant explain how this requirement is conveyed to the COL applicants and to identify the COL action item, inspections, tests, analyses, and acceptance criteria (ITAAC), or other interface requirement that addresses this.

In its response dated February 24, 2009, the applicant submitted a proposed markup to Section 3.7.5.2 of the DCD, requiring COL applications to include in their post-earthquake a procedure to check the gaps between racks and between the racks and walls, and to take appropriate actions to restore the design-basis gaps. The staff found this acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.4 “Assumptions” was revised in TR-54, Revision 2, to state that “Modeling the total effect of n individual fuel assemblies rattling inside the storage cells in a horizontal plane as one lumped mass at each of five levels in the fuel rack is a conservative assumption. Thus, the effects of chaotic fuel mass movement are incorporated into the analysis by introducing a fuel ratio factor of 0.75 (75 percent of the fuel weight is used in the analysis).”

The staff noted that the use of a 0.75 fuel ratio factor is a departure from prior revisions of TR-54, where a fuel ratio factor of 1.0 was assumed. In RAI-SRP9.1.2-SEB1-05, the staff requested the applicant provide a detailed technical basis for utilizing a fuel ratio factor of 0.75.

During the August 2009 audit, the staff and the applicant discussed this issue in depth. The applicant agreed to conduct additional analysis to justify a fuel ratio factor less than 1.0. In a letter dated November 11, 2009, the applicant reported that it could not justify a fuel ratio factor less than 1.0. Consequently, this approach to reducing the seismic loads was abandoned. The applicant implemented a number of rack design changes to demonstrate adequacy for the loads based on a 1.0 fuel ratio factor. The staff confirmed that in TR-54, Revision 3, reference to fuel ratio factor was deleted. Therefore, RAI-SRP9.1.2-SEB1-05 was resolved.

During the review of TR-54, Revision 2, the staff noted that Section 2.8.1.4, “Rack-to-Rack and Rack-to-Wall Impacts,” was revised, and indicated that the re-analysis of the spent fuel racks, to incorporate the updated seismic loading and revisions in the design of the racks, resulted in two rack-to-wall impacts: in Run 5, rack A1 impacts the west wall at a force of 45,690 lb; and in Run 4, rack B4 impacts the north wall at a force of 67,800 lb. In RAI-SRP9.1.2-SEB1-06, the staff requested the applicant describe in detail how these additional impact loads had been considered in the design of the fuel pool structure (including the liner) and in the design of the fuel racks, and also to identify where this would be described in the AP1000 DCD.

In a letter dated June 12, 2009, the applicant submitted its initial response to RAI-SRP9.1.2-SEB1-06. The staff and the applicant discussed the response at the August 2009 audit, and concluded that the rack-to-rack and rack-to-wall impact loads could increase, depending on the final resolution of the fuel ratio factor issue discussed in RAI-SRP9.1.2-SEB1-05 above. In a letter dated November 11, 2009, the applicant submitted a revised response, documenting the increased impact loads and describing the analysis method used to evaluate cell wall buckling at the top of the rack for the worst case impact load. The applicant stated that the details of the analysis were included in TR-54, Revision 3 (November 2009). The staff reviewed TR-54, Revision 3. The applicant conducted a nonlinear analysis using the LS-DYNA computer code; the results showed that the required safety factor of 1.5 is achieved before failure of the cell wall. The staff determined that the applicant’s analysis constituted an acceptable method to check the adequacy of the spent fuel rack design for the worst case top impact load. During the June 2010 audit, the staff audited the Westinghouse/HOLTEC calculation for the impact analysis, and discussed the results with the applicant. The staff found that the calculation is consistent with the information in TR-54, Revision 3, and is acceptable. The applicant also identified several DCD changes to describe the impact analysis.

In the November 11, 2009 RAI response, the applicant also stated that the impact load between the rack and the SFP wall increased from 36,741 kg (81,000 lb) to 163,293 kg (360,000 lb), but that this had only a marginal effect on the required steel liner thickness. The staff noted that it would be necessary to audit the applicable calculations, before it could accept this result. The staff attempted to audit the applicable calculations at the June 2010 fuel rack audit; however, the applicant's staff was unable to answer staff questions on the calculations.

At the DCD Section 3.8 regulatory audit conducted during the week of June 28, 2010, the staff again attempted to audit the AP1000 calculations that evaluate the spent fuel rack impact forces on the SFP walls. Again, the applicant was not able to address the questions raised by the staff.

Therefore, the staff requested the applicant to augment its prior RAI response to address the following: (1) describe how the tri-axial state of stress in the impacted faceplate has been addressed in design check, when considering the impact load in addition to other concurrent loadings; and (2) provide a comparison between the load combination with seismic load only and the load combination with seismic load and impact load, in order to confirm that the impact load is insignificant.

At the structural issues regulatory audit conducted August 18-20, 2010, the staff and the applicant discussed the results of the applicant's analyses to address the staff's questions, and the applicant's initial draft of the RAI response. The staff requested several additions to the RAI response, to which the applicant agreed. In a letter dated August 25, 2010, the applicant formally submitted its revised response, which included: (1) the calculation for the third principal stress in the faceplate of the spent fuel wall, due to the rack impact load on the wall, and a comparison of the stress intensities with and without the third principal stress, at several locations in the face plate; and (2) a comparison of the element member forces at several critical locations on the SFP wall between the load combination with seismic load only and the load combination with seismic load and impact load.

The staff reviewed the revised response and found it acceptable because the calculation results demonstrate that: (1) the effect of the increased impact force from a spent fuel rack onto the SFP walls is insignificant, for the design of SFP wall; and (2) the design of the spent fuel wall still meets the specified acceptance criteria when the impact load is included with other concurrent loads. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.8.4.1, "Cell Wall Buckling Evaluation," was revised in TR-54, Revision 2. A different buckling equation and different boundary conditions are indicated. The rectangular flat plate model representing the lower cell wall region is now assumed to be clamped on all 4 edges. Even with the assumption of clamped on all 4 edges; a very small safety margin against buckling is indicated in Revision 2. The staff determined that only one edge can truly be treated as clamped, and the other 3 edges can rotate somewhat due to the flexibility of the adjacent sections.

In RAI-SRP9.1.2-SEB1-07, the staff requested the applicant: (1) provide the technical basis for changing the boundary conditions to clamped on all four edges; and (2) identify the minimum acceptable factor of safety and the technical basis for its selection.

In a letter dated April 19, 2009, the applicant submitted its response to RAI-SRP9.1.2-SEB1-07, in which it provided its technical basis for the revised calculation of buckling for the cell wall. The staff reviewed the response and determined that the information provided was insufficient, and that a significantly expanded technical basis would be needed before the staff could accept the cell wall buckling calculation. At the August 2009 audit, HOLTEC informed the staff that it was conducting a detailed nonlinear analysis of the bottom of the rack for vertical compressive load, and presented the ANSYS computer model and preliminary results. The staff found this to be a considerable analytical improvement.

In a letter dated November 11, 2009, the applicant submitted a revised RAI response, indicating it had re-evaluated the buckling capacity of the spent fuel storage rack cells at the base of the rack using an ANSYS finite element analysis. The results show that the spent fuel rack cells remain in a stable configuration when subjected to 1.5 times the maximum seismic load without any gross yielding of the storage cell; therefore, the ASME Code requirements for Level D conditions in this area are satisfied. The ANSYS analysis and results were included in TR-54, Revision 3 (November, 2009).

During the June 2010 audit, the staff reviewed HOLTEC's final results of the ANSYS buckling evaluation of the cell walls, at the base of the spent fuel rack. The calculation shows that a 1.5 factor of safety, in accordance with the acceptance criterion in ASME Code Section III, Subsection NF, has been achieved. The staff found the analytical method used and the results obtained to be acceptable, based on its detailed review of HOLTEC's calculation. Therefore, RAI-SRP9.1.2-SEB1-07 is resolved.

In addition to the SFP rack and pool liner, the staff also performed a review of the integrity of the spent fuel assemblies during a design basis seismic event. NUREG-0800 Section 3.8.4, Appendix D, provides guidance for evaluating the consequences of seismic loads on the fuel assemblies. The applicant described its evaluation of the spent fuel rack design (including fuel assemblies) in TR-54, "Spent Fuel Storage Racks Structural/Seismic Analysis," Revision 4. The applicant calculated the maximum deceleration of a fuel assembly due to lateral SFP seismic demands as 3.38 g, assuming that all assemblies move in phase. The worst-case maximum deceleration of a fuel assembly due to out-of-phase motion, fuel-to-fuel impacts, or fuel-to-wall impacts was reported as 5.69 g. This seismic demand was less than the reported Westinghouse minimum allowable grid impact capacity of 8.92 g and results in a factor of safety of approximately 1.57.

The staff's review of TR-54, Revision 4, found that the technical basis for the minimum allowable grid impact capacity (8.92 g) was not sufficiently described in the report. To address this, the staff performed an audit on July 14, 2011, to review TR-54 supporting documentation. The staff reviewed the applicant's grid crush strength test results summarized in report NFRE-07-08, "AP1000 RFA-2 Mid Grid Dynamic Crush Strength Preliminary Verification Report," which was used to develop the AP1000 minimum allowable grid impact force. The staff confirmed that the tests were performed in accordance with previously accepted practices and that the resultant grid crush data supported the stated minimum allowable grid impact force presented in TR-54.

The staff ensured that the applicant identified the limiting failure mechanism by reviewing two research reports pertaining to spent fuel cladding integrity developed by Sandia National Laboratory (SNL) and Lawrence Livermore National Laboratory (LLNL). The staff's review of these reports (ML042710347 and ML061990439, respectively) found that spent fuel cladding can withstand forces (accelerations) of up to 10 times the grid loading limits proposed by the applicant. The staff could not, however, make conclusions with respect to grid integrity from the data within these reports, as both reports were focused on cladding ruptures and did not investigate buckling of the grids. However, data presented in the reports related to cladding were sufficient when combined with the information provided in TR-54 for the staff to conclude that grid buckling was the limiting failure mechanism for the AP1000 fuel design.

Based on the following review findings: (a) the applicant's fuel grid testing was performed in accordance with NUREG-0800 Section 4.2; (b) significant margin in the fuel assembly grid (factor of safety of 1.57); and (c) indication that grid failure is the limiting fuel assembly failure mode, the staff concludes that there is reasonable assurance that the AP1000 spent fuel assemblies, when stored in the spent fuel racks in the SFP, will retain their structural integrity when subjected to SSE level demands.

9.1.2.2.1.1.3 Conclusion

The staff has conducted a detailed review of TR-54, which addresses DCD Revision 15 COL Information Item 9.1-3: "Perform a confirmatory structural dynamic and stress analysis for the spent fuel rack, as described in Subsection 9.1.2.2.1. This includes reconciliation of loads imposed by the spent fuel rack on the SFP structure described in Subsection 3.8.4." The staff finds the spent fuel rack design, as described in TR-54, Revision 4, to be acceptable. On the basis of its review, the staff concludes that the substance of the COL information item is completely addressed by TR-54, and that completion of this COL information item is no longer needed.

In its previous evaluation of AP1000 DCD, Section 9.1.2, the staff identified that the spent fuel rack design must meet the relevant requirements in GDC 2 and GDC 4. Based on its review, the staff has concluded that the spent fuel rack design meets these 10 CFR Part 50 requirements.

9.1.2.2.1.2 Spent Fuel Rack Density

In the AP1000 DCD, Revision 17, the applicant increased the SFP storage rack density from high density to higher density racks. GDC 61 requires in part that fuel storage systems be designed with a residual heat removal capability having reliability that reflects the importance to safety of decay heat removal. As indicated in NUREG-0800 Section 9.1.2, III.1, the staff considers the design of high-density fuel storage systems to be acceptable in this regard if (among other things) low-density storage is used, at a minimum, for the most recently discharged fuel to enhance the capability to cool it. The applicant's fuel storage system design does not use low-density racks as specified by NUREG-0800 Section 9.1.2.III.1, and this difference between the proposed design and the staff's acceptance criteria has not been explained and justified. In RAI-SRP9.1.2-SBPA-14, the staff requested that the applicant address this difference and explain how the proposed fuel storage system design is adequate for satisfying GDC 61 requirements commensurate with the staff's review criteria. The staff requested that the AP1000 DCD be revised to include this information. The staff identified this as Open Item OI-SRP9.1.2-SBPA-14.

In a letter dated September 22, 2009, the applicant stated that in the previously approved DCD revision, the SFP uses only high density racks. In Revision 17 of the DCD, the applicant further increased the SFP density in order to increase storage capacity. The applicant clarified that the SFP cooling system is designed to remove the decay heat produced by the spent fuel assemblies during all modes of plant operation, regardless of the spent fuel assemblies' storage locations in the pool.

The staff evaluated the applicant's response and determined that using a SFP cooling system design with sufficient cooing capability to maintain the stored fuel cooled is an acceptable method of meeting the requirements of GDC 61. Therefore, the staff finds that the applicant's response is acceptable and the staff's concerns discussed in Open Item OI-SRP9.1.2-SBPA-14 are resolved.

The staff's detailed evaluation of the SFP cooling system capacity is documented in Section 9.1.3 of this report

9.1.2.2.2 Spent Fuel Pool Water Level Increase

The applicant proposed a series of changes related to an increase in normal SFP water level in DCD Section 9.1.2.2. The bases for these changes are addressed in TR-121. These changes include:

1. Increase the normal water volume of the pool from 685,159 to 721,121 liters (181,000 to 190,500 gallons).

2. Raise the water level from 76 cm (30 in) below the operating deck to 38 cm (15 in) below the operating deck.

3. Delete reference to "a minimum of 10 feet of shielding water above the spent fuel assemblies" and replace with "a minimum of 8.75 feet of shielding water above the active fuel height of spent fuel assemblies."

In describing spent fuel transfer operation, AP1000 DCD, Revision 15, stated that waterways are of sufficient depth to maintain "a minimum of 10 feet of shielding water above the active fuel height." AP1000 DCD, Revision 17, proposes to change Section 9.1.2.2 to state that waterways are of sufficient depth to maintain "a minimum of 8.75 feet of shielding water above the active fuel height." This corresponds to a decrease in minimum shielding of 38.10 cm (15 in) in minimum shielding from DCD Revision 15.

Also, in the AP1000 DCD, Revision 17, Tier 1, Table 2.1.1-1, "Inspections, Tests, Analysis and Acceptance Criteria," line 5, the applicant proposes to change the maximum elevation to which the bottom of a fuel bundle can be lifted from 7.70 meters (m) (25 ft, 3 in) below the operating deck to 7.47 m (24 ft, 6 in) below the operating deck. This corresponds to an additional lift of 22.86 cm (9 in).

In AP1000 DCD, Revision 17, Table 9.1-2, "Spent Fuel Pool Cooling and Purification System Design Parameters," the applicant proposes to change the SFP normal water level from 30.48 cm (12 in) below the operating deck to 38.10 cm (15 in) below the operating deck. This corresponds to a decrease in normal water level of 7.62 cm (3 in).

With the increased fuel bundle lift of 22.86 cm (9 in) and a decreased normal SFP water level of 7.62 cm (3 in), the staff believes that the change in minimum water shielding is be a decrease of 30.48 cm (12 in), not a decrease of 22.86 cm (9 in).

In RAI-SRP9.1.2-SBPA-09, the staff requested the applicant to clarify the proposed changes described above in the DCD so that the decrease in minimum shielding can be accurately determined.

On June 25, 2009, the staff conducted a regulatory audit of the DCD Revision 17 documentation and met with the applicant's personnel to identify the specific information required in order to resolve this RAI and other Chapter 9 RAIs.

During the June 25, 2009 audit, the applicant stated that the change in minimum shielding from 2.89 m (9.5 ft) to 2.67 m (8.75 ft) does not exceed the limit of 2.5 millirem to the bridge operator. The justification for this change is documented in calculation APP-GW-N2C and is discussed in the applicant's response to RAI-SRP12.3-CHPB-02.

With respect to the maximum elevation to which the bottom of the fuel bundle can be lifted, the applicant identified an ITAAC limit for mechanical hard stops. The mechanical hard stops provide 2.59 m (102 in) of water shielding.

The staff's evaluation of the justification for the minimum shielding change, as discussed in the applicant response to RAI-SRP12.3-CHPB-02, was reviewed and documented in Section 12.3 of this report. The staff identified this as Open Item OI-SRP9.1.2-SBPA-09.

In a response dated September 18, 2009, the applicant stated that DCD Revision 17, Tier 1, Table 2.1.1-1, Paragraph 5, limits the fuel assembly raise height to 7.4 m (24 ft, 6 in) between the bottom nozzle and the operating floor; elevation 41.15 m (135 ft, 3 in). This corresponds to 2.6 m (8.5 ft) of water shielding above the active portion of the fuel when the refueling cavity/SFP water level is at the 40.84 m (134 ft) elevation (minimum water elevation for fuel movement). This limit is established as a mechanical hard stop limit for the refueling machine (RM) and the FHM.

DCD Revision 17, Tier 2, throughout Section 9.1 and Section 12 establishes the minimum water coverage as 2.7 m (8.75 ft), 32 m (105 ft) elevation, above the active portion of the fuel when the refueling cavity/SFP water level is at the 40.84 m (134 ft) elevation. This is a 0.23 m (9 in) reduction in shielding. The RM and FHM would have controls to limit the hoist up travel to satisfy this requirement. Section 12.3 of this report evaluates the impact of this change on radiation protection considerations.

The staff finds that the RAI response discussed above clearly shows how the minimum shielding has been impacted by the changes in the RM limit set points. The staff determined that, based on the above discussion, Open Item OI-SRP9.1.2-SBPA-09 is resolved. The remaining changes are reviewed in Section 9.1.3 and 12.3 of this report. The staff determined that these changes made to DCD Section 9.1.2 are conforming changes and do not impact the staff's safety evaluation of DCD Section 9.1.2. Therefore, the staff finds the proposed changes acceptable.

9.1.2.2.3 Fuel Handling Crane Change

The applicant proposed to delete references to the fuel handling jib crane and replace them with references to the new-fuel handling crane in DCD Section 9.1.2.2.1. However, in response to RAI-SRP9.1.4-SBPB-01, the applicant stated in a letter dated June 26, 2008, that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated. The evaluation of this change is reviewed in Section 9.1.4 of this report. The staff determined that this change made to DCD Section 9.1.2 is a conforming change that does not impact the staff's safety evaluation of DCD Section 9.1.2. Therefore, the staff finds the proposed change acceptable.

9.1.2.2.4 Spent Fuel Criticality Analysis

9.1.2.2.4.1 Summary of Technical information

In the certified DCD Revision 15, Section 9.1.2, "Fuel Storage and Handling," it is stated in Section 9.1.6 that the COL applicant is responsible for a confirmatory criticality analysis for the spent fuel rack, as described in Section 9.1.2.3. In DCD Revision 17, the applicant proposed to change this COL action by performing the confirmatory criticality analysis so that the COL action item is no longer necessary. DCD Section 9.1.2.3 is revised to reflect that the criticality analysis is now complete, and Section 9.1.6 is revised to state that the COL information requested in this section has been completely addressed in TR-65, Revision 2, and the applicable changes are incorporated into the DCD. The applicant stated that no additional work is required by the COL applicant. The technical details of the criticality analysis for the AP1000 spent fuel storage design is presented in TR-65, Revision 2. This report provides the technical support for the changes found in Section 9.1.2 of DCD Revision 17. The staff's review of the criticality analysis of AP1000 spent fuel storage includes DCD Revision 17, Section 9.1.2 and the supporting TR-65, Revision 2.

In a letter dated September 16, 2009, the applicant stated that it will be submitting an alternate loading pattern with a restriction, which will preclude the need for using burnup credit in the Region 2 rack criticality analysis. This will result in a change to the technical specifications (TS). Open Item OI-SRP9.1.1-SRSB-08 was created to track all changes related to this restricted loading pattern and the corresponding analysis.

Subsequently, in a letter dated July 28, 2010, the applicant retracted the September 16, 2009 proposal that suggested a restricted loading pattern and clarified the applicant's intent to rely on the analysis presented in TR-65, Revision 2, with full loading as the basis for the AP1000 Revision 17 SFP criticality analysis. Furthermore, the applicant provided a response to Open Item OI-SRP9.1.1-SRSB-08 in the July 28, 2010, letter that demonstrates consistency with the burnup credit methodology used and approved in current reactors. The staff has reviewed TR-65, Revision 2, and the RAI response from the July 28, 2010, letter against NUREG-0800 Section 9.1.1, the guidance in the August 19, 1998, NRC memorandum authored by Larry Kopp "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (the Kopp guidance) and past precedence examples. The staff concludes that the applicant follows the current available guidance and therefore the response and burnup credit analysis are acceptable. Therefore, Open Item OI-SRP9.1.1-SRSB-08 is satisfied and closed.

The AP1000 design includes facilities for the onsite storage of irradiated spent fuel. The spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area.

Irradiated fuel is stored in a stainless steel-lined concrete SFP containing 8 racks capable of holding a total of 889 assemblies. The spent fuel racks are divided into two regions within the common pool. The Region 1 rack consists of stainless steel cells with neutron absorbing material (Metamic®) attached to all sides facing other storage locations. Cells are separated by a water gap. There is no absorber on the cell faces adjacent to the pool wall or when SFP geometry does not require neutron absorber panels to remain subcritical. Metamic® panels are located on Region 1 edges in locations that could physically hold an assembly between the Region 1 and 2 racks or between the Region 1 rack and pool wall. Region 2 racks consist of the same stainless steel box structure as Region 1 with fixed neutron absorber attached to the outside of the walls, but there is no intervening water gap in this case. In addition, there are five damaged fuel locations, which are of the same design as the Region 1 storage racks.

The spent fuel storage facilities are designed to meet seismic Category I requirements and maintain a subcritical storage configuration of fuel during normal storage and accident conditions.

The applicant has provided a system description in Section 9.1.2 of the DCD. In addition, TR-65, Revision 2, is reviewed as part of the AP1000 DC application. The criticality analysis is summarized here, in part, as follows:

Criticality analyses are performed for the AP1000 spent fuel storage racks to demonstrate K_{eff} ≤.95 during normal conditions, assuming a maximum nominal initial enrichment of 4.95 w% U-235 and taking into consideration uncertainties due to fuel and rack manufacturing tolerances. In addition, the spent fuel storage racks will remain subcritical under optimum moderation conditions. TR-65, Revision 2, provides the criticality analyses, including a description of the analytical methods used in the criticality analyses, as well as a description of the analytical uncertainties, equipment manufacturing tolerances, and other analysis assumptions.

In DCD Tier 1 Table 2.1.1-1, ITAAC Item 7 addresses criticality control during normal operation, design basis seismic events, and design basis dropped fuel assembly accidents.

COL Information Item 9.1-4

In Revision 17 of the AP1000 DCD, the applicant proposed to resolve COL Information Item 9.1-4 by performing a confirmatory criticality analysis for the spent fuel racks. The applicant submitted TR-65, Revision 0, for the staff's review to demonstrate that it had met the requirements of COL Information Item 9.1-4. COL Information Item 9.1-4 in the DCD is also discussed in NUREG-1793 as COL Action Item 9.1.6-4. This evaluation is a secondary review for COL Information Item 9.1-4 with respect to the compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment.

In Revision 15, Section 9.1.6 to the AP1000 DCD, COL Information Item 9.1-4 states:

> The Combined License applicant is responsible for a confirmatory criticality analysis for the spent fuel racks, as described in Subsection 9.1.2.3. This analysis should address the degradation of integral neutron absorbing material in the spent fuel pool storage racks as identified in GL-96-04, and assess the integral neutron absorbing material capability to maintain a 5 percent subcriticality margin.

In Revision 17 of the AP1000 DCD, the applicant proposed to resolve the COL information item with the following:

> The Combined License information requested in this subsection has been completely addressed in TR-65, and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant.

9.1.2.2.4.2 Evaluation

The staff's review of AP1000 DCD Section 9.1.2 follows the procedures outlined in NUREG-0800 Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3. Compliance with regulatory requirements was verified based on the criteria defined by GDC 62 and 10 CFR 50.68, "Criticality accident requirements."

The TS identified in NUREG-0800 Section 9.1.1, Section III, are reviewed in the applicable sections of this report.

Design Bases

DCD Tier 2, Section 4.3.2.6.1 is consistent with TR-65, Revision 2, in stating that the K_{eff} of fully loaded spent fuel storage racks will not exceed 0.95, assuming that the racks are flooded with potential moderator. DCD Tier 2, Section 4.3.2.6.1 also states that the maximum new fuel enrichment level must be less than or equal to 5.0 w% U-235. This is not consistent with the maximum nominal initial enrichment of 4.95 w% U-235 stated in TR-65, Revision 2.

Revision 17 of the DCD states that criticality analyses will demonstrate that the fuel storage rack geometry in combination with the integral neutron absorber material is sufficient to maintain the fuel in a subcritical condition as given above. TR-65 states that the applicant will follow the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.17, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors," with regard to criticality safety.

By following the guidelines in NUREG-0800 Section 9.1.1, the staff finds that the design bases described above for the fuel storage and handling systems meet the requirements of GDC 62 and 10 CFR 50.68(b).

Criticality Analysis Methodology

The primary method for determining the multiplication factor for the various configurations being considered in this analysis is the Monte Carlo Code MCNP-4a, with the attached nuclear data libraries ENDF/B-V and ENDF/B-VI. The applicant validated this combination against relevant critical experiments as defined in the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) International Handbook of Evaluated Criticality Safety Benchmark Experiments. This validation is used to determine the inherent bias implicit in this approach, which is added to ensure that the multiplication factor is below the acceptable limit. The MCNP calculations used an appropriate number of cycles, histories per cycle, and skipped cycles to ensure that the MCNP calculation was converged.

The design criteria are consistent with the requirements outlined in 10 CFR 50.68(b)(4) for spent fuel racks. The requirement states that: for fully loaded spent fuel racks the multiplication factor must be below 0.95 including all uncertainties and biases, and taking credit for soluble boron in the cooling water. In addition, the multiplication factor must be less than 1.0 when the fully loaded rack is flooded with fresh water. In all cases rack cells are assumed to be loaded with fuel of maximum reactivity. These criteria are used in conjunction with the latest evaluation techniques described in ANSI/ANS-8.17-2004, which can be summarized in the following relationship:

$$k_{eff} = k_c + \Delta k_p + \Delta k_c$$

Where

k_{eff} = maximum multiplication factor.
k_c = calculated multiplication factor determined by MCNP.
Δk_p = allowance for convergence, modeling, and manufacturing limitations.
Δk_c = bias uncertainty associated with code validation.

The allowance for convergence and manufacturing tolerances included the statistical accuracy of the MCNP calculations, rack tolerances, fuel tolerances, and depletion uncertainties. The code bias uncertainties are based on the comparisons with critical experiments.

The MCNP calculations are generally carried out on infinite arrays of fuel cells. This is achieved by assuming a single unit cell with reflecting or periodic boundary conditions.

The methodology presented in TR-65, Revision 2 is consistent with standard industry practice and with past NRC approved techniques. The staff, therefore, approves of the criticality analysis methodology as described in TR-65, Revision 2.

Assumptions

In Section 4 of TR-65, the applicant listed the modeling assumptions used in the analysis to ensure a conservative approach. The staff reviewed these assumptions to ensure that they maximized the k_{eff} calculations and would therefore be conservative. The staff concluded that all but one of the assumptions was conservative. The remaining assumption involved ignoring certain minor components, which the applicant claimed would in effect remove the neutron absorption by these components that would normally occur. While the staff agrees with this statement, the staff also notes that by ignoring the components in the modeling, the applicant is in effect replacing them with borated water which would also act as an absorber. The net effect is not specified; however, based on previous experience the staff feels that the change to k_{eff} based on this assumption would be insignificant and more than bounded by the other assumptions. Therefore, the staff approves of the approximations used in this calculation based on the preceeding discussion.

Input Data

The applicant used the basic input data required for the calculations as described in TR-65, Revision 2. This data covers geometric input for the racks and fuel assemblies, core operating history, burnable poison treatment and axial burnup distribution.

The staff found that borosilicate glass burnable absorber rods were not considered in the analysis. In RAI-SRP9.1.1-SRSB-07, the staff requested that the applicant describe how the use of borosilicate glass burnable absorbers affects the analysis. In its response dated September 29, 2009, the applicant stated that TR-65 was revised to include borosilicate glass burnable absorbers. The revised TR shows that they are bounded by other combinations of inserts and, therefore, have no impact on the results. The staff finds this response acceptable.

Computer Codes

The two computer codes used in this analysis are MCNP-4a and CASMO-4. They are used to determine the multiplication factors, and the core depletion behavior and sensitivity to manufacturing tolerances, respectively.

MCNP-4a

1) Appendix A of TR-65 presents a series of critical experiments that were analyzed to determine the bias and uncertainty associated with the use of MCNP-4a and its attached nuclear data library. In addition, the KENO5a Monte Carlo code and its nuclear data library are used to analyze the same experiments as a cross check of the calculated results

The staff reviewed the experiments chosen for this comparison and concludes that they include the magnitudes of the most important parameters of the fuel assemblies to be stored in the racks. The resulting values of the multiplication factor as a function of energy of the average lethargy causing fission (EALF) was analyzed using linear regression analysis. The result showed no strong trends, since the correlation coefficients are low. These calculated values for bias and uncertainties are used in calculating the maximum multiplication factor for the storage racks.

The applicant investigated the possibility of the multiplication factor having a systematic trend. The staff reviewed this investigation and agrees with the applicant's conclusion that the multiplication factor showed no discernable trend with the parameters investigated. In addition to the investigation of potential trends in the multiplication factor, the applicant investigated the sensitivity of several parameters on the multiplication factor. The staff concludes that the use of these systematic trend and parametric study investigations follows the guidance to account for all biases and uncertainties provided in the NRC memo authored by Larry Kopp, which is currently used by the industry and previously accepted by the agency.

The applicant analyzed a small number of critical experiments that included Mixed Oxide (MOX) fuel. In this manner the ability of the combination of MCNP-4a and its attached nuclear data library to handle MOX fuel could be estimated. The staff reviewed these comparisons of critical experiments to the results of the applicant's analysis. It is noted that there is a discrepancy introduced by ^{241}Pu decay and the implied growth of ^{241}Am; however, the staff concludes that this discrepancy does not appreciably affect the results. The staff concludes that the applicant's methodology provides overall conservative results and is therefore acceptable.

Based on the staff's review of the benchmark calculations presented in Appendix A of TR-65, Revision 2 as detailed in the preceding discussion, the staff finds the use of MCNP-4a and the calculated MCNP-4a bias and bias uncertainty values to be acceptable.

CASMO-4

The applicant uses CASMO-4 for the following two purposes in this analysis:

1) Depletion and decay calculations to determine the isotopic content of spent fuel, and
2) Determinations of changes in the multiplication factor introduced by perturbations in the storage rack.

CASMO-4 is an industry standard transport theory depletion code originally developed for core analyses. It has been used for SFP depletion calculations previously and approved by NRC staff. In Appendix B of TR-65, Revision 2, the applicant provides benchmarking results for CASMO-4 including reactor critical comparisons, cross section library comparisons, and code to code comparisons.

> The applicant compared the k-inf values calculated with CASMO-4 to experimental reactor critical data. This database covered a wide range of design conditions to ensure the calculated uncertainties would be applicable to the AP1000 design.
>
> The applicant investigated the effects of using different available cross-section data libraries. The results did not indicate a significant difference between the investigated libraries.
>
> The applicant provided a code to code reactivity difference comparison with MCNP-4a as an additional check on top of the critical experiments comparisons. These comparisons showed good agreement and did not result in differences that required additional scrutiny.

The staff reviewed the methodology of the depletion code benchmarking against the Kopp memo as well as other recently approved license amendment request applications. The staff reached the following conclusions regarding the depletion code benchmarking:

(1) The applicant compared the calculated results to a variety of relevant critical experiments. While it is recognized that more recent relevant data have been made available, the experiments used by the applicant provide reasonable assurance that the relevant biases can be determined. In RAI-SRP9.1.1-SRSB-06, the staff requested additional information regarding the effects of performing the analyses at maximum water density (4 °C) and the lack of Tungsten gray rod data in the benchmark tests. In response, the applicant explained that the extrapolation from the minimum temperature used in the analysis to 4 °C results in a negligible effect on the uncertainty calculations. The staff determined that while this effect should normally be quantified, the additional analytical margin included on top of the calculated bias uncertainty is sufficient to bound the small effect that would result from slightly more dense water. The applicant addressed the question regarding lack of Tungsten gray rod data by stating that the effect of tungsten inserts would not appreciably affect the benchmark studies which already include various other absorber inserts. The staff agrees that this observation is most likely correct, but notes that there currently is no database to support this conclusion as this is a new material. The staff approves the use of the applicant's methodology based on the wide range of other parameters investigated, but recognizes that small future changes could occur based on the collection of new data.

(2) Based on the comparison of the cross-section libraries, the staff finds that the library selection between current industry standard libraries has little effect on the reactivity calculation. Therefore the staff finds that the library used by the applicant is acceptable.

(3) The staff confirmed that applicable permutations were investigated to determine the appropriate depletion uncertainties.

(4) The applicant included an additional analytical margin on top of the calculated bias uncertainty.

Therefore, the staff approves the use of CASMO-4 for the depletion calculations necessary for the AP1000 SFP criticality analysis.

Criticality Analysis

In this section the review of the criticality calculations for the Region 1 and Region 2 racks from TR-65, Revision 2 is presented. In addition, the review of the consequences due to possible abnormal and accident scenarios is discussed.

The Region 1 and Region 2 racks are represented by detailed 2-D MCNP models, with reflecting boundary conditions on the surfaces separating one cell from the other. In this manner, an infinite array of fuel cells is represented in the calculation. Additionally, MCNP models that explicitly involve more storage cells are used to analyze the abnormal or accident conditions.

The 3-D MCNP calculations have the same detail in the X-Y plane as the 2-D models, and are extended axially in the Z-direction. The model includes a 30 cm axial water reflector, which does not include any boron, even for those cases that include boron in the water in the storage cells.

The CASMO calculations are 2-D, and thus the regions above and below the fuel are not represented. CASMO is used to determine the perturbations in the multiplication factor due to manufacturing tolerances. These perturbations are presented as adjustment factors to the multiplication factor determined for a nominal fuel loading case.

Region 1 Storage Racks

Region 1 storage racks are qualified to store fresh fuel with an enrichment of up to 4.95 w% U-235. There are 243 Region 1 storage locations in the storage pool. The geometric representation is as described previously in this report.

As part of the Region 1 criticality analysis, the applicant analyzed various abnormal conditions in addition to the standard loading. These included eccentric fuel assembly positioning, uncertainties due to manufacturing tolerances, water temperature/density, and accident conditions.

The staff reviewed the criticality analyses for Region 1 as presented against the guidance in NUREG-0800 Section 9.1.1. As part of a regulatory audit held May 6-7, 2009, the staff inspected the computer runs used by the applicant to ensure that the methodology was correctly followed. Based on the review of the methodology and its application, the staff finds the Region 1 racks to be acceptable for use as presented in TR-65.

Region 2 Storage Racks

The applicant performed the Region 2 SFP criticality analysis in a manner similar to the Region 1 analysis, except that the analysis included the use of burnup credit in order to meet the requirements of 10 CFR 50.68. The applicant followed the guidelines provided by the Kopp memo as well as the methodology of recently approved license amendment requests in calculating the burnup credit.

In RAI-SRP9.1.1-SRSB-08, NRC staff questioned the applicant's burnup credit assumption that a 5 percent reactivity uncertainty penalty included the effects of missing nuclide data on the computational biases and uncertainties. In response to this RAI, the applicant's September 16, 2009, letter described a loading pattern restriction on the Region 2 racks and its plan to submit a simplified analysis that does not require burnup credit. This plan will not require any changes to the physical rack design as presented in TR-65. Subsequently, after SFP license amendment requests that included similar burnup credit methods were accepted by the agency, the applicant submitted a letter dated July 28, 2010, requesting that the agency return to reviewing the full-capacity SFP criticality design in TR-65 Revision 2.

The staff compared the applicant's burnup credit methodology to the guidance provided in the Kopp memo as well as to SERs for recently approved SFP license amendment requests. The staff notes that the burnup credit guidance does not explicitly state how an applicant should handle biases and bias uncertainties and whether or not the 5 percent reactivity uncertainty covers them when considering potential lack of data regarding specific isotopes.

As a result of reviewing the recent SERs, the staff determined that the submittals had been based on approaches to the implementation of burnup credit similar to that used by the applicant in TR-65, Revision 2. The staff determined that past precedent supports the applicant's position that the use of the 5 percent reactivity uncertainty penalty has been approved previously to cover the depletion bias uncertainty. The staff considers Open Item OI-SRP9.1.1-SRSB-08 closed.

During the regulatory audit held May 6-7, 2009, the staff reviewed the depletion and criticality calculations used by the applicant in the AP1000 Region 2 SFP criticality analysis. The staff determined that the applicant used the codes previously approved in this section and correctly applied the applicable code biases and bias uncertainties while calculating the values for k_{eff}. Based on the staff's review of the methodology and analysis, as detailed in this section, along with the precedent set by the recent approval of applications using burnup credit, the staff concludes that the applicant's analysis of the Region 2 SFP criticality demonstrates compliance with the requirements of 10 CFR 50.68 by following the Kopp memo.

Revised Rack Dimensions

DCD Revision 18 introduced an increased thickness of the Region 1, Region 2 and damaged fuel storage rack cell walls, for the purposes of increased structural stability. DCD Revision 19 includes additional changes related to this change in thickness. In a letter dated March 3, 2011, the applicant submitted TR-65 Revision 3. Revised criticality analyses based on the latest spent fuel pool rack design are located in Supplement 1 of TR-65, Revision 3.

Following the same methodology as found in TR-65 Revision 2, the new analyses provide new k_{eff} calculations based on the burnup loading curve found in TS 3.7.12 (Figure 3.7.12-1) of the DCD. These new calculations demonstrate that the k_{eff} values are lower for all of the analyzed fuel as compared with the previous analysis (TR-65, Revision 2), except for the fuel with 2.5 wt% and 4.0 wt% enrichment. The calculations of k_{eff} for fuel with 2.5 wt% and 4.0 wt% enrichment demonstrate an increase, but this increase is within the administrative margin maintained by the applicant in the previous analysis (TR-65, Revision 2). These increased k_{eff} values do not violate the limits specified in 10 CFR 50.68.

The staff reviewed the calculations included in Supplement 1 of TR-65, Revision 3 and confirmed that the same methodology presented in Revision 2 was used including the determination and use of biases and uncertainties. The revised k_{eff} values do not violate the regulatory limits. Therefore, staff finds the AP1000 spent fuel pool criticality calculations based on TR-65, Revision 3 acceptable.

Restrictions and Limitations

The AP1000 SFP design as presented in TR-65, Revision 2 is approved for use with the following limitation:

> Limitation #1: Applicability
>
> The AP1000 SFP is approved for storing the fuel types presented in (or bounded by) TR-65, Revision 2. Any fuel not bounded by those used in TR-65, Revision 2 (higher enrichment, different burnable absorbers designs than analyzed, etc.) will require further analysis.

<u>COL Information Item 9.1-4 Evaluation</u>

To assure compliance with GDC 4, NUREG-0800 Section 9.1.2 III.2.G states the reviewer should verify that "the materials wetted in the SFP, (e.g., spent fuel racks, fixed neutron poison, and the SFP liner) and, if applicable, the new fuel vault are chemically compatible and stable. The review also verifies whether there are potential mechanisms to alter the dispersion of any strong fixed neutron absorbers. The secondary reviewer provides input for this review." NUREG-0800 Section 9.1.2 I.11.B further states that "the reactivity of fuel in the SFP is controlled by plates or inserts attached to spent fuel racks containing neutron poison dispersed in a matrix. In some environments, the matrix may degrade and release the neutron poison, resulting in some reduction of neutron absorbing properties of the panels. The licensee should have a program for monitoring the effectiveness of the neutron poison present in the neutron absorbing panels."

The staff has reviewed the information included in TR-65, which identifies the neutron absorber material in the spent fuel storage racks as Metamic®, a metal matrix composite material consisting of a Type 6061 aluminum (Al 6061) alloy matrix reinforced with boron carbide (B_4C). TR-65, Section 2.4.8 describes testing to qualify the Metamic® material for spent fuel rack service, including short and long-term elevated temperature tests, accelerated corrosion and radiation tests, mechanical properties and neutron transmission testing. The staff has previously issued an SER approving a topical report supporting the use of the Metamic® material in spent fuel racks in an operating plant. The operating plant subsequently submitted a license amendment request to use the Metamic® material in the SFP, which was approved via an SER issued by the NRC staff. The staff noted that the same generic vendor report supporting the application to use Metamic® in the operating plant is referenced in TR-65. The SER for the license amendment at the operating plant placed conditions on the use of the Metamic® material: specifically, implementation of a coupon sampling program to ensure performance consistent with the laboratory qualification testing.

The Metamic® absorber material is relied upon in the TR-65 criticality analysis to maintain the required 5 percent subcriticality margin. While TR-65, Section 2.4.8 indicates no significant loss of neutron absorbing capacity is expected for the Metamic® material based on the testing conducted, the Metamic® material is a new material with very little operating experience in the SFP environment. Spent fuel racks with Metamic® have been installed at the operating plant but the time in service for these racks as of March 2008 has been only a few months, and no coupons have been withdrawn or tested. TR-65 and the DCD include no mention of the coupon surveillance program implemented by the operating plant, nor do they recommend a similar program for the AP1000 plants.

Although the data from the operating plant surveillance program could be used to confirm the laboratory test results and could be extrapolated to the Metamic® in the AP1000, a relatively small amount of data from the operating plant will be available when construction begins for the first AP1000 plants. Further, the service conditions for the Metamic® material in the operating plant may not be identical to the expected service conditions for the Metamic® material in the AP1000 design. Additionally, some qualification tests such as the radiation testing only encompassed a 40 year rather than a 60 year life. Therefore, the staff considered that a coupon sampling plan similar to that implemented in the operating plant should be implemented for the AP1000 plants. Therefore, in RAI-SRP9.1.2-CIB-01, the staff requested that the applicant provide the following information, and include the information in the next revision to the AP1000 DCD:

1) A description of the neutron absorbing material to be used in the spent fuel storage racks. The description should include the material type, chemical composition, and mechanical properties, and a discussion of the suitability of the absorber material for long-term use in the SFP environment. Include a description of any testing performed to qualify the material for 60 years service in the SFP environment, specifically with respect to corrosion and radiation degradation. The description should also address whether the absorber material has an anodized finish, the anodizing process used, and the cleaning process to ensure removal of surface contaminants prior to installation.

2) A description of the recommended program to be implemented by the licensee to confirm that the behavior of the neutron absorbing material is consistent with the behavior of the material in the qualification tests. For example, the DCD may need to identify a COL item requiring the COL applicant to include a description in the COL application of the coupon sampling or monitoring program for the licensee to implement when the plant is placed into commercial operation.

The applicant responded by letter dated April 18, 2008. With regard to question #1, the applicant stated that the material that will be used in the AP1000 fuel storage racks is Metamic, a metal matrix composite material consisting of a Type 6061 aluminum alloy matrix reinforced with boron carbide (B_4C) as described in TR-65. The Metamic® will be in the form of sheets having a nominal thickness of 0.269 cm (0.106 in) and a minimum ^{10}B areal density of 0.0304 gm/cm^2 (minimum 30.5 wt percent B_4C). The panels are not intended to be anodized, but will be cleaned via glass bead blasting and washing with demineralized water to ensure removal of surface contamination prior to installation. The applicant also included the density, yield and ultimate strength, and elongation of the material in its description of the material.

In the April 18, 2008 letter, the applicant also described the testing performed to qualify the Metamic® material for 60 years service in the SFP environment. The applicant referenced proprietary testing by a vendor (HOLTEC) as documented in HOLTEC Report HI-2043215, "Source Book for Metamic® Performance Assessment," Revision 2, HOLTEC International, dated September 2006 (not publicly available) (hereafter referred to as the "HOLTEC report") as its basis for qualification of the material:

- Elevated temperature testing of 31 w% B_4C Metamic® at 398.9 °C (750 °F) in air for nearly a year. There was no reduction in thickness, change in weight, reduction in ^{10}B

content or change in density. The applicant stated that the results of this test demonstrate that exposure to a temperature of 48.9 °C (120 °F) in the SFP will not detrimentally affect the condition of the Metamic® panels.

- Accelerated corrosion testing at 93.30 °C (200 °F) for 90 days. No corrosion was observed and no significant change of ^{10}B areal density. The applicant stated that while these tests were carried out at a temperature only 26.7 °C (80 °F) higher than the typical upper bound, this is sufficient to yield results representative of longer periods. The applicant referenced Department of Energy (DOE) fundamentals Handbook DOE-HDBK-1015/1-93 Module 2 – Corrosion, which states "A temperature rise in the range of -6.7 °C to 10 °C (20 °F to 50 °F) doubles the corrosion rate until the formation of the protective oxide film is complete." The applicant also stated that the aluminum oxide layer that forms on the Metamic® is largely inert and that once the protective oxide film forms the corrosion rate becomes approximately zero. The applicant also referenced the DOE handbook with regard to the effect of pH on corrosion rate. The handbook indicates essentially zero corrosion rate at pH 5.5 and a corrosion rate of nearly zero in a pH range of 4-8. The applicant stated that the normal pH of the AP1000 SFP is within this range. The applicant stated that the complete lack of any chemical changes in these tests, combined with the knowledge of the effects of temperature and pH on corrosion rate, is sufficient to show that the aqueous pool environment, even for 60 years or more, will not detrimentally affect the condition of the Metamic® Panels Radiation testing of 31 w% B_4C with both gamma (1.5 x 10^9 Gy (1.5 x 10^{11} rads)) and fast neutron (1.7 x 10^{18} to 5.8 x 10^{19} n/cm^2) components. The conclusions of the post irradiation testing were that the Metamic® exhibited excellent dimensional stability after irradiation, and there was no change in Boron-10 areal density.

In response to the second part of the RAI, the applicant stated that an in-situ surveillance program to monitor the condition of the Metamic® in the racks will be implemented for the AP1000 spent fuel racks. The program uses representative material coupons, and is patterned after similar programs used for years at operating plants. The specific Metamic® monitoring program will be developed by the COL applicant. The applicant recommended the following tests to be performed on the coupons:

1) Neutron attenuation measurements (to verify the continued presence of boron)

    a. Acceptance criteria – A decrease of no more than 5 percent in Boron-10 content, as determined by neutron attenuation, is acceptable.

2) Thickness measurement (as a monitor of potential swelling)

    a. Acceptance criteria – An increase of thickness at any point should not exceed 10 percent of the initial thickness at that point

The applicant also included a markup for DCD Section 9.1.2.2.1 including a description of the Metamic® material (including all the information described above) and the qualification testing performed. The applicant also included a markup for a new COL Information Item 9.1.6.7 in Section 9.1.6 of the DCD, which reads as follows:

> The COL holder shall implement a spent fuel rack Metamic® coupon sampling or monitoring program when the plant is placed into commercial operation.

The staff reviewed the response to RAI-SRP9.1.2-CIB-01 and finds that the applicant provided an adequate description of the material.

The topical report SER for the operating plant placed conditions upon the use of the material; specifically, that a coupon surveillance program be implemented. The coupon surveillance program was to include the following attributes:

- size and types of coupons to be used (i.e., similar in fabrication and layout to the proposed insert including welds and proximity to stainless steel);
- technique for measuring the initial B_4C content of the coupons;
- simulation of scratches on the coupons;
- frequency of coupon sampling and its justification; and tests to be performed on coupons (e.g., weight measurement, measurement of dimensions (length, width and thickness), and B_4C content); these tests should also address, at a minimum, any bubbling, blistering, cracking, flaking, or areal density changes of the coupons, any dose changes to the coupons, or the effects of any fluid movement and temperature fluctuations of the pool water.

The coupon surveillance program approved for the operating reactor included visual examination and photography, measurement of weight and density, and measurement of the length and width dimensions in addition to thickness.

The acceptance criterion proposed by the applicant for ^{10}B content is a decrease of no more than 5 percent as determined by neutron attenuation, which is essentially the same as the acceptance criteria approved for the operating plant of any change in ^{10}B content of 5 percent. The staff finds the applicant's proposed criterion for ^{10}B content acceptable since it is consistent with that previously approved by the staff.

The acceptance criterion for thickness for the operating plant is any change in thickness of (+/-) 0.025 cm (0.01 in) for a 0.25 cm (0.1 in) coupon. The applicant's proposed acceptance criterion for thickness is no increase in thickness greater than 10 percent. The staff finds that the applicant's criterion is acceptable because it will detect significant swelling of the material thickness.

As part of the material qualification, the applicant also cited the results of radiation testing of 31 wt percent B_4C with both gamma (1.5×10^9 Gy (1.5×10^{11} rads)) and fast neutron (1.7×10^{18} to 5.8×10^{19} n/cm^2) components. An appendix to the HOLTEC report indicates that the gamma dose of 1.5×10^9 Gy (1.5×10^{11} rads) is roughly equivalent to the exposure Metamic® would receive in 40 years of actual fuel rack service. Although the AP1000 plant design life is 60 years, the staff finds the use of a gamma dose equivalent to 40 years exposure acceptable since the plant license duration will be 40 years, and the recommended monitoring program includes testing to verify the continued capability of the Metamic® materials to provide the required neutron absorption capacity. The HOLTEC report did not compare the fast neutron exposure expected in fuel pool service to the fast neutron exposure in the qualification program. However, the coupon monitoring program recommended by the applicant is likely to detect any degradation associated with fast neutron exposure.

The staff finds the referenced corrosion testing is appropriate. The accelerated corrosion testing resulted in essentially no corrosion of the material. Aluminum and aluminum alloys form a passive oxide film in most air or water environments that limits general corrosion to negligible rates. In the corrosion testing, some Metamic® coupons with a mill finish experienced pitting corrosion. The topical report summarized the results of a corrosion test program performed by Electrical Power Research Institute (EPRI). The duration of the EPRI corrosion tests was slightly over one year. The pitting in the EPRI tests was attributed to impurities present on the coupon surface based on the fact that the coupons cleaned by glass beading (as the AP1000 Metamic® material will be cleaned) or chemically cleaned prior to anodizing did not experience pitting. The corrosion testing performed by EPRI was verified by tests documented in the HOLTEC report performed in similar environments, but for a shorter duration (90 days). The applicant concluded the accelerated test results are sufficient to show that the aqueous SFP environment, even for 60 years, will not detrimentally affect the condition of the Metamic® panels. However, the applicant did not provide a quantitative basis for extrapolating the corrosion test results to 60 years. Although the staff agrees that corrosion appears to have been stopped by the formation of a passive film, due to the limited experience with Metamic® in operating reactors, the staff did not agree that a corrosion concern can be completely precluded for Metamic.

In the response to RAI-SRP9.1.2 CIB1-01, the applicant described the mounting and location of the coupons in the SFP, but did not provide the size. The applicant indicated that the coupons would be precharacterized for weight, dimensions (especially thickness) and ^{10}B loading, but did not provide: the technique for measuring the initial B_4C content; a recommended schedule for withdrawal and testing of the coupons; whether coupons included scratches; recommended tests to address bubbling, blistering, cracking, flaking, or areal density changes of the coupons; any dose changes to the coupons; or the effects of any fluid movement and temperature fluctuations of the pool water.

Since the applicant did not provide recommended criteria for several of the items addressed in the conditions on the use of Metamic® in the SER, and due to the limited experience with Metamic® material in operating reactors, particularly with regard to long-term corrosion behavior, in an April 28, 2008 supplement to RAI-SRP9.1.2-CIB1-01, the staff requested the following additional information:

> Provide a recommendation to the COL applicant for the following aspects of the Metamic® coupon surveillance program, and include the same information in the next revision to the DCD:
>
> - recommended coupon withdrawal schedule
> - size and types of coupons to be used (i.e., similar in fabrication and layout as the proposed insert including welds and proximity to stainless steel);
> - technique for measuring the initial B4C content of the coupons;
> - whether the coupons should include simulated scratches, or explain why simulated scratches are unnecessary.
> - tests to monitor bubbling, blistering, cracking, or flaking.

- test to monitor for corrosion, such as weight loss measurements and/or visual examination.

If any of these items are not recommended, provide a justification for excluding the item from the program.

In response to the supplementary request, the applicant submitted a revised response to RAI-SRP9.1.2-CIB1-01, dated June 20, 2008. The response to the supplementary request stated that the applicant and the COL applicants together are providing a Metamic® coupon surveillance program. The applicant is responsible for the design aspects of the Metamic® coupon surveillance program and the COL applicants are responsible for the programmatic aspects.

The supplemental response also stated the following items should be included in the site surveillance program, and indicated whether the applicant or the COL applicant was responsible for each item as follows:

- Recommended coupon withdrawal schedule-the applicant
- Size and types of coupons to be used (i.e., similar in fabrication and layout as the proposed insert including welds and proximity to stainless steel)- the applicant
- Technique for measuring the initial B4C content of the coupons-COL applicants
- Whether the coupons should include simulated scratches, or explain why simulated scratches are unnecessary-the applicant
- Tests to monitor bubbling, blistering, cracking, or flaking-COL applicants
- Test to monitor for corrosion, such as weight loss measurements and/or visual examination-COL applicants

The supplemental response further stated that this information is described in the COL holder's Metamic® coupon surveillance program, and that the applicant has worked with HOLTEC to design the Metamic® Coupon Tree requiring eight coupons for 60 years of surveillance. Based on this, the applicant has specified a coupon tree with 14 coupons (six additional coupons). The applicant provided a revision of the DCD markup from the original response to RAI-SRP9.1.2-CIB1-01 describing the number and size of the coupons, and showing the recommended coupon withdrawal schedule. The staff finds the information provided on the sizes and types of coupons is acceptable because the coupons are cut from the actual Metamic® absorber panels; therefore, the coupons are representative of the actual absorber panels, including the presence of any scratches. Therefore, scratches will not be deliberately added to the coupons. The staff also finds the proposed coupon surveillance schedule acceptable because it requires more frequent testing of the coupons early in plant life when problems are more likely to be detected, and covers the entire 60-year design life of the plant. The applicant also included a markup of DCD Table 1.8-2 showing COL Information Item 9.1-7 for a Metamic® Monitoring Program (also included in the original RAI response).
Section 9.1.6.7 of the proposed DCD markup included the text of the COL information item:

> The COL holder shall implement a spent fuel rack Metamic® coupon sampling or monitoring program when the plant is placed into commercial operation.

However, the staff found this wording did not include a sufficient level of detail to provide direction to the COL applicant with respect to the Metamic® Monitoring Program elements, as described in the supplemental RAI response.

The staff held a telephone conference with the applicant on July 11, 2008, to clarify whether the DCD would be revised to incorporate all the information provided in the supplemental response, and clarify the requirements for the Metamic® Monitoring Program. During the teleconference, the applicant agreed to provide a revised supplemental response that would include the text for COL Information Item 9.1-7 describing the elements of the Metamic® Monitoring Program for which the COL applicant is responsible. The revised supplemental response was received via letter dated August 21, 2008. In the revised response, the applicant provided a markup of DCD Section 9.1.6.7, which added details of the Metamic® Monitoring Program. The additional information stated that this program would include tests to monitor bubbling, blistering, cracking, or flaking and a test to monitor for corrosion such as weight loss measurements and or visual examination. However, the tests listed did not include the two tests originally proposed by the applicant in its original April 18, 2008, response to RAI-SRP9.1.2-CIB1-01, namely neutron attenuation and thickness tests.

In a response dated April 21, 2009, the applicant provided a revised markup of DCD Section 9.1.6.7 that includes the neutron attenuation and thickness tests in the COL information item text, in addition to those tests previously identified. The staff, therefore, considers this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The portion of COL Information Item 9.1-4 that addresses compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment meets GDC 4. Therefore, the staff finds that the DCD changes regarding this issue, as proposed by the applicant in TR-65, are acceptable.

9.1.2.2.4.3 Conclusion

The staff has reviewed the AP1000 SFP criticality analysis and methodology as presented in TR-65 Revision 2 and concludes that the AP1000 SFP design is acceptable for spent fuel storage as described in the application and with the limitations as listed in this safety evaluation.

The staff finds the applicant’s proposed resolution to AP1000 COL Information Item 9.1-4, which addresses the compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment, meets GDC 4 and is, therefore, acceptable. Furthermore, the staff finds that the TR-65 conclusions regarding the evaluation for compatibility of the neutron absorbing materials used in the spent fuel racks with the spent fuel environment are generic and are expected to apply to all COL applications referencing the AP1000 DC.

9.1.2.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of AP1000 DCD, Section 9.1.2, the staff identified acceptance criteria based on the design's meeting relevant requirements in GDC 2, GDC 4, GDC 5, GDC 61, GDC 62, and in GDC 63. The staff found that the AP1000 spent fuel pool cooling system (SFS) design was in compliance with these requirements of NUREG-0800 Section 9.1.2 and determined that the design of the AP1000 spent fuel storage, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 spent fuel storage as documented in AP1000 DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 9.1.1, and 9.1.2. The staff finds that the applicant's proposed changes do not affect the ability of the spent fuel storage to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. The staff concludes that the AP1000 new fuel storage design continues to meet all applicable acceptance criteria. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. The proposed changes to the spent fuel rack design and criticality analysis contribute to the increased standardization of the certification information in the AP1000 DCD and, thus, meet the requirements of 10 CFR 52.63(a)(1)(vii). Therefore, the staff finds that the proposed changes to AP1000 Section 9.1.2 are acceptable.

9.1.3 SFP Cooling and Purification

9.1.3.1 Summary of Technical Information

Section 9.1.3, "SFP Cooling System," of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revision 17, the applicant proposed the following changes to Section 9.1.3 of the certified design:

1. The applicant proposed to increase the SFS pumps' common suction pipe diameter from 15.24 cm (6 in) to 25.4 cm (10 in) from the SFP to the penetration at the SFS pump room and then reduced from 25.4 cm (10 in) to 20.32 cm (8 in). Where the common suction pipe branches off to the individual SFS pumps, the pipe is reduced again, to 15.24 cm (6 in). The applicant documented this change in TR-103, "Fluid System Changes," APP-GW-GLN-019, Revision 2 of October 2007.

2. The applicant proposed to increase the SFS pumps' common discharge pipe diameter from 15.24 cm (6 in) to 20.32 cm (8 in). The applicant documented this change in TR-103.

3. The applicant proposed to increase the number of spent fuel storage locations in the SFP from 619 fuel assemblies to 889 fuel assemblies. The applicant documented this change in TR-103.

In the AP1000 DCD, Revision 17, the applicant submitted changes documented in its response to RAI-TR103-SBPA-01 on November 9, 2007. The additional changes included the following:

a. The applicant modified Section 9.1.3.1.3.1, "Partial Core," for the assumed heat load to be based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the SFP, including 44 percent of a core (69 assemblies) placed in the pool beginning at 120 hours after shutdown.

b. The applicant modified Section 9.1.3.1.3.2, "Full Core Off-Load," for the assumed heat load to be based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the SFP, plus one full core placed in the pool at 120 hours after shutdown.

c. The applicant modified Table 9.1-2, "SFP Cooling and Purification System Design Parameters," to reflect the new SFP storage capacity of 889 fuel assemblies.

d. The applicant modified Section 9.1.3.4.3, "Abnormal Conditions," with regard to the decay heat levels in the SFP and the amount of makeup water required to provide fuel pool cooling in the event of an extended loss of normal SFP cooling. The applicant also reduced the lengths of time when no makeup is needed and when safety-related makeup from the cask washdown area is sufficient to achieve SFP cooling from 7 days to 72 hours.

e. The applicant modified Table 9.1-4, "Station Blackout/Seismic Event Times," with regard to the event descriptions, time to saturation, and height of water above fuel at 72 hours and at 7 days. In addition, the applicant revised note 7 for Table 9.1-4.

f. As a result of design changes to the shield building, the availability of the water in the passive containment cooling water storage tank (PCCWST) for refilling the SFP has changed. This impacted the basis of several scenarios of the SFP thermal analysis. These changes include:

    (i) The applicant modified TS, DCD Chapter 16, Section 3.6.7, "Passive Containment Cooling System (PCS) – Shutdown," to lower the required reactor decay heat limit for air-only containment cooling from 9 megawatts thermal (MWt) (30.7 million British thermal units (MBtu)) to 6 MWt (20.5 MBtu).

    (ii) The applicant modified Section 9.1.3.4.3, "Abnormal Conditions," with regard to the required safety-related makeup water sources. The cask loading pit (CLP) is now credited as a safety-related makeup water source when the PCSWST is not available to provide safety-related makeup water to the SFP and the SFP heat load is higher than 5.6 MWt (19.1 MBtu) and less than 7.2 MWt (24.6 MBtu).

(iii) The applicant modified TS 3.7.9, "Fuel Storage Pool Makeup Water Sources," to verify that the CLP is available and communicated to the SFP before it is needed as a makeup water source.

(iv) The applicant modified Section 9.1.3.4.3 with regard to required makeup water flow from the passive containment cooling ancillary water storage tank (PCCAWST) for post 72 hour cooling. The change allows the system to adjust the makeup flow between 132.5 Lpm (liters per minute (35 gallons per minute (gpm)) and 189.3 Lpm (50 gpm) as needed.

(v) The applicant modified Table 9.1-4, with regard to the event descriptions and time to saturation at 72 hours and at 7 days.

4. The applicant proposed to raise the specified maximum allowable elevation for the bottom of a spent fuel assembly to be within 7.47 m (24 ft, 6 in) of the operating deck. The applicant documented a change to maximum allowable elevation in TR-121 and a subsequent change in AP1000 DCD, Revision 17.

   The applicant proposed to revise AP1000 DCD, Tier 1, Table 2.1.1-1, "Inspections, Test, Analysis, and Acceptance Criteria," and changed the acceptance criteria for design commitment number 5, to say, "the bottom of the dummy fuel assembly cannot be raised to within 24 ft, 6 in of the operating deck floor."

   The applicant also proposed to change the normal water level in the SFP from 0.610 m (2 ft) below the operating deck to 0.381 m (15 in) below the operating deck. This change results in an increase in normal water inventory in the SFP from 685,159 liters (181,000 gallons) to 721,121 liters (190,500 gallons).

5. The applicant proposed to modify the design basis refueling boron concentration to be 2700 parts per million (ppm). The applicant documented this change in TR-18, "AP1000 Core & Fuel Design," APP-GW-GLR-059 (WCAP-16652-NP), Revision 0 of October 2006.

6. The applicant proposed to revise the description of where the main suction line for the SFP cooling system connects to the SFP from "at an elevation 0.61 m (2 ft.) below the normal water level of the pool" to "at an elevation 1.83 m (6 ft.) below the operating deck."

   The applicant also revised the description of SFP alarms in the main control room (MCR) from stating that alarm in the MCR from safety-related instrumentation occurs "when water level reaches either the high level or low level setpoint" to "when level in the SFP reaches the low-low level setpoint."

7. The applicant modified the limiting site interface air temperatures in AP1000 DCD. These changes are evaluated in Section 2.3.1, "Regional Climatology"; Section 12.2, "Ensuring that Occupational Radiation Doses Are as Low as Is Reasonably Achievable"; and Section 12.4, "Radiation Protection Design" of this report. Section 9.1.3.1.3.1 and Section 9.1.3.1.3.2 were modified to reflect these changes. Additionally, Section 9.1.3.1.3.1 and Section 9.1.3.1.3.2 were modified to clearly describe when the different temperature limits are applicable to the SFS.

9.1.3.2 Evaluation

The staff reviewed all changes to the SFS in accordance with NUREG-0800 Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." The staff reviewed all changes identified in AP1000 DCD, Revision 17. The staff did not re-review descriptions and evaluations of the SFS in AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in the applicant's TRs, RAI responses, and the DCD itself.

The regulatory basis for AP1000 DCD, Section 9.1.3, is documented in NUREG-1793. The staff has reviewed the proposed changes to DCD Section 9.1.3 against the applicable acceptance criteria of NUREG-0800 Section 9.1.3. The following evaluations discuss the results of the staff's review.

The specific criteria that apply to the proposed DCD changes are; 10 CFR 52.63(a)(1)(vi), which concerns substantially increasing overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), which concerns contribution to the increased standardization of the certification information in the AP1000 DCD.

9.1.3.2.1 SFS Pump Common Suction Pipe Diameter Increase

In Revision 17 of the AP1000 DCD the applicant proposed to increase the SFS pumps' suction pipe diameter. The basis for this change is documented in TR-103. In TR-103, Section II.B.6, the applicant states that the previously specified SFS pumps' suction pipe diameter of 15.24 cm (6 in) from the SFP to the individual pumps resulted in large pressure drops, which could cause cavitation in the SFS pump suction lines when the SFP temperature is elevated. The large suction line pressure drop created an unacceptable condition in which SFP cooling with the SFS pumps could have become incapable of restoration following a temporary loss of SFP cooling. The increase in suction line diameter will reduce the pressure drop in the suction line and increase the net positive suction head available (NPSHa) for SFS pump operation.

The staff finds that the safety function of the SFS continues to be met because the change does not affect SFP water level or makeup capability, and capability to keep the spent fuel assemblies cooled and covered with water is not affected by this change. In addition, the staff finds that the operational flexibility of the SFS pumps is increased because NPSHa will be adequate under a wider range of operating conditions. The staff finds that the system continues to comply with GDC 61 with regard to decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions and that this change is needed because it increases the NPSHa for the SFS pumps and provides increased operational flexibility to support restart of the pumps following a loss of cooling event in which the SFP temperature becomes elevated. Therefore, the staff finds the proposed change to be acceptable.

9.1.3.2.2 SFS Pump Common Discharge Pipe Diameter Increase

In TR-103, Section II.B.6, the applicant states that the increase in the SFS pump common discharge pipe diameter, combined with the increased suction pressure provided by the increased SFS pumps' common suction pipe diameter, reduces the SFS pumps' required head and allows the pumps' horsepower at normal operating conditions to be lowered by approximately 35 percent. This change provides additional operational flexibility by supporting a decrease in the required pump horsepower without degrading safety-related functions or operating margins in the SFS. The staff finds that the safety function of the SFS continues to be met because the change does not affect SFP water level or makeup capability, and capability to keep the spent fuel assemblies cooled and covered with water is not affected by this change. The staff finds that this change is acceptable because it provides additional operational flexibility for the SFS pumps, and the design continues to comply with GDC 61 with regard to decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions. The staff noted that there was an AP1000 documentation discrepancy because this change, which is described in TR-103, Revision 2, was not reflected in AP1000 DCD, Revision 16.

In RAI-SRP9.1.3-SBPA-02, dated April 16, 2008, the staff requested that the applicant update the application to reflect the proposed change in SFS pump common discharge pipe diameter. In its response dated May 28, 2008, the applicant stated that the increase in SFS pump common discharge pipe diameter should have been reflected in Revision 16 of the AP1000 DCD and that it would be captured in the next revision of the AP1000 DCD. In DCD Revision 17, Figure 9.1-6 “Spent Fuel Pool Cooling System Piping and Instrumentation Diagram,” the applicant proposed to change the SFS common discharge pipe diameter to 20.32 cm (8 in) as identified in the response to RAI-SRP9.1.3-SBPA-02. Therefore, the staff finds the applicant’s response to RAI-SRP9.1.3-SBPA-02 to be acceptable.

9.1.3.2.3 Increase in Number of Spent Fuel Storage Locations

In TR-103, Section II.B.17, the applicant states that the storage capacity of the spent fuel storage racks in the SFP has been updated to provide 889 spent fuel storage locations, an increase of 270 from the previous 619 locations. As a result of the increased spent fuel storage capacity, the maximum decay heat input to the SFP is increased under various refueling offload conditions. The applicant updated the SFP thermal analysis to demonstrate that the SFS can maintain the stored fuel, cooled and submerged under water for 72 hours after the initiating event from safety-related sources and up to 7 days from internal sources.

During the June 25, 2009 audit, the staff requested clarification with respect to the AP1000 DCD, Revision 17 change in Figure 9.1-4, “Spent Fuel Storage Pool Layout (889 Storage Locations).” In RAI-SRP9.1.3-SBPA-15, the staff asked the applicant to explain the inconsistency in Figure 9.1-4 between Revision 16 and 17 in that Rack C1 contained an arrangement of 12 x 10 (minus 7 cells) assemblies in Revision 16 and 12 x 10 (minus 2 cells) assemblies in Revision 17.

In its response dated August 25, 2009, the applicant described that the Revision 16 Rack C1 label was incorrect and was corrected in Revision 17 to be arranged in a 12 x 10 module, with 2 cells missing in the North-South direction. The staff determined that this change was an editorial correction and not a design change. The rack description provided in the DCD is consistent with this correction. Based on the above discussion, the staff finds the applicant’s response to RAI-SRP9.1.3-SBPA-15 to be acceptable and the issue is resolved.

In Revision 17 of the DCD the applicant updated Tier 2, Table 9.1-4 to reflect the calculated height of water above the fuel at 72 hours and at 7 days after the seismic event, for the three limiting offload scenarios. Table 9.1-4 contained a number of notes. Note 6 stated:

> Alignment of the PCS ancillary water storage tank and initiation of PCS recirculation pumps provide a makeup water supply to maintain this pool level or higher above the top of the fuel.

In Revision 15 of the DCD, this note only applied to SFP cooling for the period of time between 72 hours and 7 days. AP1000 DCD, Revision 17 added this note to the first offload scenario described in Table 9.1-4 for the period of time prior to 72 hours.

The staff determined that this change was inconsistent with the system description provided in the TS Basis for TS 3.7.9, and inconsistent with the staff's position documented in NUREG-1793 for AP600, AP1000 Revision15, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs"; and SECY-98-161, "The Westinghouse AP600 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems."

In Revision 1 to Open Item OI-SRP9.1.3-SBPA-13, the staff requested that the applicant clarify/justify if the AP1000 design is in accordance with the established staff position, or if the design is being changed to introduce a new design basis.

In its response dated August 20, 2010, the applicant stated that PCCAWST was never credited for SFP makeup prior to 72 hours, and the addition of Note 6 to the height of water above the fuel prior to 72 hours was an editorial error. The applicant's response also included a markup of Table 9.1-4 removing Note 6 from the height of water above the fuel prior to 72 hours.

Based on the applicant's response, the staff found the applicant's response acceptable and Revision 1 of Open Item OI-SRP9.1.3-SBPA-13 is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In Revision 15 of the DCD Tier 2 Section 9.1.3.1.3.1, the applicant stated that the assumed partial core heat load is based on the decay heat generated by the accumulated fuel assemblies stored in the fuel pool for 10 years plus 44 percent of a core (68 assemblies) being placed into the pool. In Revision 17 of the DCD Tier 2 Section 9.1.3.1.3.1, the applicant states that the assumed partial core heat load is based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the fuel pool, which includes 44 percent of a core (69 assemblies) being placed into the pool. Additionally, in Revision 15 of the DCD Tier 2 Section 9.1.3.1.3.2, the applicant stated that the assumed full core offload heat load is based on the decay heat generated by the accumulated fuel assemblies stored in the fuel pool for 10 years, plus one full core placed in the pool. In Revision 17 of the DCD Tier 2 Section 9.1.3.1.3.2, the applicant states that the assumed full core offload heat load is based on the decay heat generated by the accumulated maximum number of fuel assemblies stored in the fuel pool, plus one full core placed in the pool.

The staff reviewed these changes and determined that by assuming that all SFP locations are filled, the calculated heat load bounds the worst possible offload scenario. The staff found that these changes were consistent with the increase in SFP capacity, were conservative in nature, and resulted in more limiting conditions. Therefore, the staff finds the proposed change to increase the total number of fuel assemblies used to calculate the heat load in the SFP for the partial core and the full core offload scenarios acceptable.

AP1000 DCD, Revision 15, Section 9.1.3, discusses three offload scenarios that represent the bounding SFP heat loads for all anticipated accident conditions. These decay heat loads are inputs for the thermal analysis of the SFP cooling.

- The first offload scenario postulates that a seismic event (concurrent with a station blackout) occurs while the reactor is operating immediately following a 44 percent core refueling. Since the reactor is operating when the event occurs, the thermal analysis assumes that the decay heat of the reactor is higher, or equal to 9 megawatt-hours MWh (30.7 MBtu); therefore, the PCCWST is reserved for containment cooling and it cannot be credited to provide safety-related makeup water to the SFP.

- The second offload scenario postulates that a seismic event (concurrent with a station blackout) occurs after a refueling is completed, and that this refueling occurred immediately following a previous 44 percent core offload. After a refueling is completed, the decay heat in the reactor is lower than 9 MWh (30.7 MBtu); therefore, the PCCWST can be credited to provide safety-related makeup water to the SFP.

- The third offload scenario postulates that a seismic event (concurrent with a station blackout) occurs after an emergency full core offload has been completed, and that this occurred immediately following a previous 44 percent core offload. This offload scenario represents the highest possible decay heat load in the SFP. Since the reactor is assumed to be empty, the PCCWST is credited for providing safety-related makeup water to the SFP.

The AP1000 DCD Revision 15 SFP is designed to use safety-related water sources to remove the SFP decay heat for the first 72 hours following events when the normal SFP cooling system is unavailable. For all the offload scenarios discussed above, the stored fuel in the SFP remains covered with water using only safety-related makeup water sources for the first 72 hours after the onset of the event. After the first 72 hours and before 7 days, the SFP credits the use of regulatory treatment of nonsafety systems (RTNSS) to provide makeup water to the SFP. The minimum water level necessary to achieve sufficient cooling of the stored fuel is the subcooled, collapsed water level (without vapor voids) required to cover the top of the fuel assemblies.

In AP1000 DCD Revision 15, the thermal analysis credits the water volume in the SFP (below the minimum water inventory level), the fuel transfer canal (including gate volume), the cask wash-down pit, and the PCCWST as the safety-related makeup water sources available for the first 72 hours of the event (depending on the offload scenario evaluated, some sources may not be available). Establishing makeup from the cask wash-down pit and the PCCWST requires operator action to re-position manual valves. The AP1000 DCD Revision 15 credits providing makeup water from the nonsafety-related water source in the PCCAWST to the SFP between 72 hours and 7 days after the event. This water source, piping segments, and the pumps are classified as RTNSS Class B in Westinghouse Commercial Atomic Power (WCAP)-15985, Revision 2, “AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process,” dated August 2003, and have availability controls.

In AP1000 DCD Revision 17, the applicant proposed several changes to Section 9.1.3 related to the thermal analysis. The basis for these changes were documented in; TR-103; TR-54; TR-65; TR-105, APP-GW-GLN-105, “Building and Structure Configuration, Layout, and General Arrangement Design Updates” Revision 2 of October 2007; and TR-108, APP-GW-GLN-108, “AP1000 Site Interface Temperature Limits,” Revision 2 of September, 2007.

The staff reviewed all of these TRs and the SFP thermal analysis report documented in APP-SFS-M3C-012, "AP1000 Spent Fuel Pool Heatup, Boiloff, and Emergency Makeup on Loss of Cooling," and determined that additional information was needed. The staff submitted RAI-TR103-SBPA-01, RAI-SRP9.1.3-SBPA-04, RAI-SRP9.1.3-SBPA-08, RAI-SRP9.1.3-SBPA-13, and RAI-SRP9.1.3-SBPA-05; requesting additional information related to the SFP thermal analysis inputs, assumptions, methodology and results. The staff's concerns related to the SFP thermal analysis inputs, assumptions, and results were identified as Open Item OI-SRP9.1.3-SBPA-04.

In response to RAI-TR103-SBPA-01 and RAI-SRP9.1.3-SBPA-04, the applicant stated that the conditions assumed for the calculated decay heat levels are the off-load conditions described in AP1000 DCD Sections 9.1.3.1.3.1 and 9.1.3.1.3.2 and that the calculated values are representative of the limiting off-load conditions as described in the applicable AP1000 DCD sections.

During the June 25, 2009 regulatory audit of the SFP thermal analysis report, the staff identified that the thermal analysis credited non-conservative assumptions related to the initial SFP water level. The staff identified this issue as Open Item OI-SRP9.1.3-SBPA-08(b), and requested that the applicant correct or justify these report findings. The applicant responded to the staff's question by revising the thermal analysis report eliminating the non-conservative assumption. On December 8, 2009, the staff performed a regulatory audit of the revised thermal analysis report and confirmed that the initial SFP water level after a seismic event had been reduced. This reduction in water level eliminated the non-conservative assumption identified by the staff in Open Item OI-SRP9.1.3-SBPA-08(b); therefore, the staff considers Open Item OI-SRP9.1.3-SBPA-08(b) closed.

The applicant also confirmed that there is no nonsafety-related piping connections in the SFP below an elevation of 39.27 m (128.83 ft), which is the minimum water level assumed in the SFP thermal analysis report. The SFP piping that extends below an elevation of 39.27 m (128.83 ft) are equipped with anti-siphon devices that prevent draining the SFP below the minimum inventory limit and are designed to be capable of performing their safety function following a design basis seismic event. During the audit of the applicant's SFP thermal analysis report, the staff verified that the analysis assumes that the initial water level is 39.27 m (128.83 ft). Therefore, the staff finds that the SFP thermal analysis was performed using assumptions in accordance with the system design and the system description provided in DCD Section 9.1.3.

The applicant subsequently revised the thermal analysis by introducing a change in methodology. The new methodology assumed that the boiling temperature of the water in the SFP would be affected by the pressure produced by the elevation of the column of water. This assumption allowed the SFP water temperature to rise above 100 °C (212 °F) before boiling. The staff determined that the applicant had not properly justified this assumption and requested that the applicant provide justification for this assumption or revise the thermal analysis calculation.

In response the applicant revised the thermal analysis removing the assumption that the SFP water temperature would rise above 100 °C (212 °F) before boiling. On January 25, 2010, the staff performed a regulatory audit of this revised thermal analysis and confirmed that this assumption had been removed. Therefore, the staff finds this acceptable, since the thermal analysis no longer changes the previously approved methodology.

Since the staff audits confirmed that the revised SFP thermal analysis used an approved methodology and used conservative inputs and assumptions, the staff considers Open Item OI-SRP9.1.3-SBPA-04 resolved.

In Revision 17 of the DCD the applicant updated Tier 2, Table 9.1-4, “Station Blackout/Seismic Event Times,” to reflect the results of the revised thermal analysis calculation. The table showed that the time to boil for all three limiting offload scenarios had decreased. For the most limiting scenario (full core offload) the time to boil decreased from 2.5 hours to 1.37 hours. In a later revision to the SFP thermal analysis report, the time to boil for this offload scenario was raised to 2.33 hours. The staff’s evaluation of this increase in time to boil is evaluated further below in this SE section.

In RAI-SRP9.1.3-SBPA-13, the staff requested that the applicant update the DCD in order to address the impact of the initial decrease in SFP time to boil on the required operator actions needed to cope with this event. The staff also pointed out that due to changes in the SFP thermal analysis, the information in Note 8 of Table 9.1-4 needed to be updated.

The applicant responded to RAI-SRP9.1.3-SBPA-13 in letters dated August 25, 2009, and February 10, 2010. The applicant’s response proposed to modify Note 8 to properly represent the revised thermal analysis. The staff found this acceptable since the DCD table is now consistent with the revised thermal analysis. The applicant’s response also stated that under most limiting conditions with the highest SFP decay heat, the operator will have more than 18 hours after boiling has begun to establish safety-related makeup. In addition, the applicant proposed to add a new Note 9 to Tier 2, Table 9.1-4 applicable to all off-loading scenarios analyzed.

The proposed Note 9 stated “operator action to align makeup water to the spent fuel pool must occur within 18 hours of the seismic event.” The staff determined that the proposed wording of Note 9 did not clearly reflect the minimum time available for operator action. This was identified as Open Item OI-SRP9.1.3-SBPA-13 in the SER with open items.

In letter dated August 6, 2010, the applicant submitted for staff review APP-GW-GLR-096 Revision 1, (Proprietary) and APP-GW-GLR-097 Revision 1 (Non-Proprietary), "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses." In a letter dated March 10, 2011, the applicant submitted for staff review APP GW GLR 096 Revision 2, (Proprietary) and APP GW GLR 097 Revision 1 (Non Proprietary). These letters also included markups of the DCD sections impacted by this change. These changes to the Shield Building impacted the availability of the safety-related water source in the PCCWST to be used as makeup to the SFP for two of the offload scenarios discussed in the DCD. The evaluation of the shield building design changes are evaluated in Section 6.2 of this report. This section evaluates the impact of these changes on the SFS.

The DCD markups included a proposed revision to Note 9 of Tier 2, Table 9.1 4. The revised note states that “[a] minimum of 18 hours is available for operator action to align makeup water to the spent fuel pool after a seismic event.” The revised Note 9 clearly states the minimum time that the operator has to perform the required actions to align safety-related makeup water to the SFP. The staff finds that the applicant’s calculation demonstrates that the operator will have sufficient time to take the required actions to align the SFP makeup sources to prevent the SFP boildown to a water level that would uncover the stored fuel. Therefore, the staff concerns identified in Open Item OI-SRP9.1.3-SBPA-13 are considered closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

AP1000 DCD, Revision 15, TS 3.6.7, “Passive Containment Cooling System (PCS) - Shutdown,” required that while the reactor decay heat is at or higher than 9 MWh (30.7 MBtu) , the PCCWST is reserved for containment cooling. When the decay heat in the reactor is below 9 MWh (30.7 MBtu), the PCCWST is credited for providing SFP safety-related water makeup. As a result of the design changes to the shield building described in the August 6, 2010 letter, the PCCWST is now reserved for containment cooling while the reactor decay heat is at or higher than 6 MWh (20.5 MBtu).

AP1000 DCD, Revision 15, TS 3.7.9, “Fuel Storage Pool Makeup Water Sources,” states that the PCCWST is required in order to increase the SFP heat load above 5.4 MWh (18.4 MBtu). The PCCWST is not available for both containment cooling and SFP makeup simultaneously. Once enough fuel has been transferred to the SFP to raise the decay heat in the SFP to 5.4 MWh (18.4 MBtu), the PCCWST needs to be available as a safety-related source for SFP makeup. However, during normal refueling operations, the reactor decay heat would still be greater than 6 MWh (20.5 MBtu); therefore, the PCCWST would not be available for SFP makeup since it is still required as safety-related source for containment cooling. TS 3.7.9 prevents additional fuel offloading until the PCCWST can be credited to provide safety-related makeup water to the SFP. In order to reduce the impact of this change on the refueling schedule, the applicant proposed to credit the CLP as a safety-related makeup water source.

The applicant revised the SFP thermal analysis to take credit for the water stored in the CLP as a safety-related makeup water source. While revising the SFP thermal analysis, the applicant identified an error in the calculation. The calculation erroneously assumed that at the onset of the initiating event, the SFP already had received 1 hour of decay heat. This error affects all of the calculated times to boil. On July 16, 2010, the staff audited the most recent and the previous thermal analysis calculation and confirmed that previous thermal analysis had also included this error. The applicant also identified that even after the PCCWST is required to be available for the SFP makeup, the PCCWST isolation valves (V001A/B/C) can still be automatically actuated. If the PCCWST isolation valves are opened the water would be drained onto containment instead of being sent to the SFP. The operators have 24 hours to take action to close the valves until Surveillance Requirement (SR) 3.7.9.1 is violated (the PCCWST volume is drained to < 1.514×10^6 liters (L) (400,000 gallons)). To prevent inadvertent actuation of these valves, the applicant proposed to modify SR 3.7.9.1 to ensure that one motor-operated valve (MOV) isolation valve (gate valve) is closed and secured prior to the PCCWST becoming operable for SFP makeup. The PCCWST air-operated isolation valves (V001A/B) cannot be used because they are fail-open. During a loss of onsite and offsite power the valves will lose compressed air and eventually open.

The staff evaluated the justification and the markups included in APP-GW-GLR-096, Revision 1, and determined that this proposed change to SR 3.7.9.1 provides assurance that the safety-related makeup water volume needed for SFP cooling would be available when needed. The staff evaluated APP GW GLR 096, Revision 2 and determined that this revision did not introduce any new changes to the information discussed above. Therefore, the staff finds the proposed change to SR 3.7.9.1 acceptable.

The revised thermal analysis concluded that the CLP contains sufficient water to allow the SFP decay heat limit to be raised 7.2 MWh (24.6 MBtu). The report also identified that if the accident scenario were to occur during a refueling outage, when the reactor decay heat is higher than 6

MWh (20.5 MBtu) and the SFP heat load is below 7.2 MWh (24.6 MBtu), the PCCASWT flow rate limits will not be sufficient for maintaining the stored fuel covered. DCD Section 6.2.2.4.2, “Preoperational Testing,” and Section 9.1.3.4.3, “Abnormal Conditions,” states that the PCCASWT has the capability of providing a total of 378 Lpm (100 gpm) to the PCCWST for containment cooling while providing 132 Lpm (35 gpm) to the SFP for SFP cooling. With the higher SFP heat load, the boiloff rate at 72 hours is higher and the rate of water makeup is also higher.

APP-GW-GLR-097, Revision 1 described the scenario in which the SFP heat load is at its highest, and that this peak is not coincident with the peak demand for containment cooling. The PCCAWST provides SFP water makeup and containment cooling, simultaneously, for the period of time between 72 hours and 7 days after the onset of the event. The applicant proposed to revise DCD Sections 6.2 and 9.1.3 to specify that the makeup flow from the PCCAWST will be throttle/adjusted to provide 303 Lpm (80 gpm) to the PCCWST and over 189 Lpm (50 gpm) to the SFP when additional flow is required in the SFP. The applicant stated that these new flow rates provide sufficient cooling for both the containment and the SFP cooling.

The staff confirmed that a flow rate of 189 Lpm (50 gpm) is higher than the anticipated boiloff rate at 72 hours (for this offload scenario). The staff also verified the proposed update to DCD Section 9.1.3.4.3 presented in the DCD markups and determined that the new description of the system operation is in accordance with the new design. Therefore, the change was found to be acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant also proposed to modify TS 3.7.9 to reflect the new thermal analysis assumptions. The modified TS require the CWP to be operable when the SFP decay heat load is higher than or equal to 4.6 MWh (15.7 MBtu) and less than 7.2 MWh (24.6 MBtu). The change adds the requirement of having the CLP available when the SFP decay heat load is higher than 5.6 MWh (19.1 MBtu) and less than or equal to 7.2 MWh (24.6 MBtu). The PCCWST is required as SFP makeup water source if the SFP decay heat is higher than 7.2 MWh (24.6 MBtu) and the reactor decay heat load is less than 6 MWh (20.5 MBtu). The applicant proposed to modify DCD Section 9.1.3, TS surveillance requirements, and the TS basis to reflect these changes. The staff identified an apparent inconsistency between TS 3.7.9 and DCD Section 9.1.3, related to the maximum decay heat that the spent fuel pool is capable of dissipating in 72 hrs with no additional makeup. In a letter dated February 25, 2011, the applicant proposed to change all references to the maximum decay heat limit for the SFP with no makeup available from 4.6 MWh (15.7 MBtu) to 4.7 MWh (16.0 MBtu). The staff found the new limit is in accordance with the SFP thermal analysis report previously evaluated by the staff and, therefore, the proposed changes were found acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text,which resolve this issue.

Establishing makeup water flow from the CLP following the initiating event requires no operator action. The applicant proposes to establish makeup water flow by opening the gate that separates the SFP and the CLP. New SR 3.7.9.4 requires verification that the CLP water level is at or higher than 4.2 m (13.75 ft) (minimum level) and that the CLP and the SFP are in communication, prior to exceeding the new SFP decay heat load limit of 5.6 MWh (19.1 MBtu). The staff finds this proposed SR 3.7.9.4 provides assurance that the required safety-related makeup water is going to be available when needed; therefore, the staff finds the proposed

SR 3.7.9.4 acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff finds that the applicant's thermal analysis has demonstrated that, with the increased spent fuel storage capacity, sufficient water inventory and sufficient makeup capability are available to keep the spent fuel covered with water under all limiting conditions, consistent with the safety functions described in DCD Section 9.1.3.5, and in accordance with NUREG-0800 Section 9.1.3. As a result of the SFP capacity increase, the time to reach saturation and the height of water coverage over stored spent fuel has changed.

For the first offload scenario (seismic event occurred while the reactor is at power immediately following a 44 percent refueling), saturation is reached in 7.38 hours. During the first 72 hours, crediting only safety-related sources, the height of water above the fuel is maintained at 0.42 m (1.4 ft). Between 72 hours and 7 days, the applicant credits the use of the PCCAWST (a RTNSS Class B system) to provide makeup water to the SFP and maintain the height of water above the fuel at 0.42 m (1.4 ft).

For the second limiting offload scenario (seismic event occurred after refueling, immediately following a 44 percent refueling), saturation is reached in 5.59 hours. If the decay heat inside the reactor is at or higher than 6 MWh (20.5 MBtu), the PCCWST is not available to provide safety-related makeup water to the SFP, and TS 3.7.9 limits the SFP heat load to 7.2 MWh (24.6 MBtu). The remaining safety-related water sources have sufficient water inventory to cover the stored fuel for 72 hours. If the decay heat inside the reactor is below 6 MWh (20.5 MBtu), the PCCWST is available to provide safety-related makeup water to the SFP. Therefore, offloading operations can continue and the SFP decay heat load can be higher than 7.2 MWh (24.6 MBtu). During the first 72 hours, crediting only safety-related sources; the height of water above the fuel is maintained at 1.3 m (4.2 ft). Between 72 hours and 7 days, the applicant credits the use of the PCCAWST to provide makeup water to the SFP and maintain the height of water above the fuel at 1.3 m (4.2 ft).

For the third limiting offload scenario (seismic event occurred after completing and emergency full core offload, immediately following a 44 percent refueling), saturation is reached in 2.33 hours. There is no fuel in the reactor; therefore the PCCWST is available to provide safety-related makeup water to the SFP. During the first 72 hours, crediting only safety-related sources, the height of water above the fuel is maintained at 2.43 m (8 ft). Between 72 hours and 7 days, the applicant credits the use of the PCCAWST to provide makeup water to the SFP and maintain the height of water above the fuel at 2.43 m (8 ft).

With the increased number of spent fuel assemblies, the staff finds that the SFP continues to maintain water coverage above the spent fuel assemblies for at least 72 hours following a loss of the nonsafety-related SFP cooling system, using only safety-related makeup water, and that adequate time is available for operators to establish the required makeup water.

The staff also finds that the SFP continues to maintain water coverage above the spent fuel assemblies for at least 7 days following a loss of the nonsafety-related SFP cooling system, using RTNSS "B" makeup water, and that adequate time is available for operators to establish makeup water from on-site sources. Because adequate cooling and coverage of spent fuel bundles is maintained, the staff finds that the SFS continues to comply with requirements of GDC 61 related to provisions for decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions. Based on the capability of the SFP and the SFS to maintain adequate cooling for spent fuel under limiting conditions and continued compliance with requirements of GDC 61, the staff finds the proposed change to credit the CLP as a safety-related makeup water source, and the increase in SFP spent fuel locations from 619 to 889 locations, acceptable.

9.1.3.2.4 Increase in Maximum Allowable Elevation of a Spent Fuel Assembly and Increase in Normal SFP Water Level

In AP1000 DCD Revision 17 the applicant proposed to increase the maximum allowable elevation of a spent fuel assembly and increase the normal SFP water level. Both Tier 1 and Tier 2 information are affected by these changes. The justification for these changes is included in TR-121.

To support the increased fuel assembly height, the applicant proposed to increase the specified normal SFP water level from 40.61 m (133.25 ft) to 40.92 m (134.25 ft). In AP1000 DCD, Section 9.1.3.1.4, "Spent Fuel Purification," the applicant revised the discussion of exposure rates to say, "The spent fuel pool cooling system is designed to limit exposure rates to personnel on the SFP fuel handling machine to less than 2.5 millirem per hour." In AP1000 DCD, Table 9.1-2, "Spent Fuel Pool Cooling and Purification System Design Parameters," the applicant updated the SFP water volume to 724,906 L (191,500 gallons), including racks without fuel, at a water level of 30.48 cm (12 in) below the operating deck. Previously, in AP1000 DCD, Revision 15, the SFP water volume was stated as 685,159 L (181,000 gallons), including racks without fuel, at a water level of 76 cm (30 in) below the operating deck.

The staff noted the applicant's statement in TR-121 that SFP water level increases .31 m (1 ft) from 40.61 m (133.25 ft) to 40.92 m (134.25 ft) is inconsistent with the applicant's statement that before the proposed change the normal SFP water level is 76 cm (30 in) below the operating deck, and after the proposed change the normal SFP water level would be 30.48 cm (12 in) below the operating deck. In RAI-SRP9.1.3-SBPA-03, dated April 16, 2008, the staff asked the applicant to clarify the amount of increase in normal SFP water level and to discuss how the water level change impacts previous analyses.

In its response dated May 28, 2008, the applicant stated that the correct SFP water level is 40.92 m (134.25 ft), and that this is consistent with a water level that is 30.48 cm (12 in) below the operating deck. The applicant stated that Revision 15 of the AP1000 DCD was inconsistent in that it stated that the water level was at an elevation 40.61 m (133.25 ft), but it also stated that the water level was 76 cm (30 in) below the operating deck. The applicant stated that TR-121 and Revision 16 of the DCD corrected this inconsistency and that all affected analyses were included in TR-121 and were performed assuming an SFP water level of 40.92 m (134.25 ft). The staff noted that in Revision 17 to the DCD, the applicant had changed the normal SFP water level from 30.48 cm (12 in) below the operating deck to 38.1 cm (15 in) below the operating deck, which correlates to a SFP water volume decrease of approximately 3785 L (1000 gallons). During the June 25, 2009 audit, the staff asked the applicant to verify that the Revision 17 change in normal SFP water volume did not impact the calculations in APP-SFS-M3C-012 and to discuss all assumptions that changed in the decay heat calculations.

The applicant clarified that the decay heat calculations in APP-SFS-M3C-012 are based on a SFP water level that correlates to the lowest non-seismic component connected to the SFP, because the assumption is that all non-seismic components attached to the SFP will fail. The lowest non-seismic component connected to the SFP is the SFS main suction line, which is below the normal operating water level.

Based on its review, the staff finds the applicant's response acceptable because the applicant corrected the inconsistency between SFP water level elevation and SFP surface location below the operating deck. On the basis that the updated values, as stated in TR-121 and Revision 17 of the AP1000 DCD, are consistent and these values have been evaluated against the decay heat analysis calculations in APP-SFS-M3C-012, the staff's concerns described in RAI-SRP9.1.3-SBPA-03 are resolved.

The staff noted that in Revision 17 to the DCD the applicant had increased the minimum combined water volume of the SFP and fuel transfer canal. In RAI-SRP9.1.3-SBPA-07 the staff asked the applicant to justify the increase in minimum combined water volume of the SFP and fuel transfer canal from 176,778 L (46,700 gallons) in Revision 16 of DCD Table 9.1-2 to 490,210 L (129,500 gallons) in Revision 17 of DCD Table 9.1-2, and to identify the effects, if any, on the decay heat calculations. During the June 25, 2009 audit, the applicant clarified that the Revision 16 value was an error and the decay heat calculations assumed the minimum combined water volume of 490,210 L (129,500 gallons) in Revision 17 of DCD Table 9.1-2. The staff verified this information and subsequently withdrew RAI-SRP9.1.3-SBPA-07.

The staff's evaluation in Section 9.1.3 of this report is limited to consideration of the operating and thermal hydraulic performance characteristics of the SFS. The proposed change to increase the maximum allowable elevation of a spent fuel assembly does not affect the operating and thermal hydraulic performance characteristics of the SFS. The heat input to the SFP and heat removal capability of the SFS is not affected by this change. The staff finds the proposed increase in maximum allowable elevation of a spent fuel assembly to be acceptable from the standpoint of effects on operation and thermal hydraulic performance of the SFS. The impact of this change on fuel handling is evaluated by the staff in Section 9.1.4 of this supplement to NUREG-1793. The impact of this change on radiation exposure of operating personnel is evaluated by the staff in Section 12.2 and Section 12.4 of this supplement to NUREG-1793.

The staff reviewed the proposed change to increase the available SFP coolant inventory and the NPSH available at the suction of the SFS pumps. The staff finds that these changes improve or enhance the previously available operating margins for the SFS pumps under normal operating conditions. Under conditions where loss of SFP cooling might occur, the staff finds that the increased water inventory in the SFP allows for a longer time before the SFP would reach saturation and provides longer times for operators to take corrective actions to reestablish SFP cooling. The staff finds that the SFS continues to comply with requirements of GDC 61, related to provisions for decay heat removal and capability to prevent reduction in fuel storage coolant inventory under accident conditions. Because operating margins are improved and the system continues to conform to requirements of GDC 61, the staff finds that the proposed increase in SFP water level is acceptable with regard to effects on the operating and thermal-hydraulic performance characteristics of the SFS.

9.1.3.2.5 Increase in Specified Design Basis Refueling Boron Concentration

In TR-18, Revision 0, the applicant proposed a change in the specified design basis refueling boron concentration, increasing the specified boron concentration from 2500 ppm to 2700 ppm in the SFP water. The applicant stated that this change is for consistency with the accumulator boron concentration value provided in the DCD Chapter 15 accident analysis. The staff noted that the accumulator boron concentration value stated in AP1000 DCD, Chapter 15 accident analysis is 2600 ppm, not 2700 ppm.

In RAI-SRP9.1.3-SBPA-01, dated April 16, 2008, the staff asked the applicant to clarify what the correct value for boron concentration is and to resolve any inconsistencies in the documentation.

In its response dated May 28, 2008, the applicant stated that the correct SFP boron concentration is 2700 ppm and that this is consistent with values stated in Section 9.1.3.2 and Table 9.1-2 of Revision 16 of the AP1000 DCD and the in-containment refueling water storage tank (IRWST) boron concentration. The applicant stated that the accumulator boron concentration in Chapter 15 is stated as 2600 ppm and that this is consistent with TS SR 3.5.1.4, which requires the boron concentration in each accumulator to be between 2600 and 2900 ppm.

In its evaluation, the staff noted that the change in SFP boron concentration from 2500 ppm to 2700 ppm is consistent with the allowable boron concentration values specified for the accumulators credited in the DCD Chapter 15 accident analysis. The staff noted that the accumulator boron concentration value stated in the DCD Chapter 15 accident analysis is 2600 ppm because that is the minimum boron concentration permitted by TS SR 3.5.1.4. The staff finds this provides a satisfactory explanation of what is meant by the statement that "this change is made for consistency with the accumulator boron concentration values." Therefore, the staff finds the applicant's response to be acceptable and the staff's concern in RAI-SRP9.1.3-SBPA-01 is resolved.

The reactivity control effects of this change are evaluated in Section 4.3 of this supplement to NUREG-1793. The effects of this change on the criticality evaluation of the stored spent fuel are evaluated in Section 9.1.2 of this report. With regard to operational and thermal-hydraulic performance of the SFP cooling system, the staff finds this change acceptable because the boron content of the SFP water has no effect on the heat input to the SFP, the heat removal capability of the SFS, the operating margins or the performance characteristics of the SFS.

9.1.3.2.6 Changes to Piping Diagrams for the Spent Fuel Pool Cooling and Safety-Related Instrumentation

The staff noted that, in the AP1000 DCD Revision 17, Figure 9.1-5, "Piping Diagrams for Spent Fuel Pool Cooling (Normal Operation)," the applicant removed branch lines shown to and from the cask pit, changed the SFP connection to chemical and volume control system (CVS) from a separate penetration to a connection shared with the SFS pump suction, deleted the return line from the SFS discharge going to the in-containment refueling water storage tank, and changed several valve types on the drawing.

In RAI-SRP9.1.3-SBPA-10, the staff requested the applicant to provide justification for the changes mentioned above and to discuss whether any of these changes impact the safety conclusions. Also, for the CVCS connection, the staff requested the applicant to clarify whether the previously used SFP penetration has been removed entirely or whether it remains as "unused and capped," which is what the revised drawing appears to indicate. This was identified as Open Item OI-SRP-9.1.3-SBPA-10 in the SER with open items.

During the June 25, 2009 audit, the applicant explained that deleting the cask pit branch lines was not a change in design. The cask pit was originally designed with a common drain line; and the drawing was corrected to represent the actual design. With respect to the CVCS connection, the change also represented a correction in the drawing, and not a design change.

The design of the CVCS never had a penetration in the SFP. It was designed to connect to the SFS pump suction line, as represented in DCD Revision 17, Figure 9.1-5.

In a letter dated September 17, 2009, the applicant stated that none of the clarifications on Figure 9.1-5 represented changes to the safety conclusions for the AP1000 SFS. These changes were introduced to correct discrepancies between Figure 9.1-6 (which represented the correct design of the SFS) and Figure 9.1-5, which included errors.

The staff reviewed the applicant's response and determined that none of the changes in Figure 9.1-5 represents a technical change; they are corrections to represent the actual SFS design as described in Section 9.1.3. The staff considers Open Item OI-SRP9.1.3-SBPA-10 resolved.

In AP1000 DCD Revision 17, Section 9.1.3.7.D, the applicant proposed to change the description of MCR alarms. In DCD Revision 15, the applicant stated that safety-related instrumentation is provided to give an alarm in the MCR when the water level in the SFP reaches either the high level or low level setpoint. In DCD Revision 17, the applicant states that safety-related instrumentation is provided to give an alarm in the MCR when the water level in the SFP reaches the low-low-level setpoint. The staff issued RAI-SRP9.1.3-SBPA-11, which reads as follows:

a) Provide the basis for the safety-related level instrumentation change,

b) Clarify whether any MCR or local alarms are available to give an alarm on high level or on low level setpoints in the SFP, as previously described in AP1000 DCD, Revision 15, and

c) Justify the impact of the change in SFP level alarms to previously performed safety evaluations or operator response evaluations.

During the June 25, 2009 audit, and in an RAI response dated October 2, 2009, the applicant clarified that only the safety-related low-low level alarm is located in the safety-related instrumentation; the high and low level setpoint signals come from nonsafety-related instrumentation and, therefore, are not referred to as safety-related instrumentation in Revision 17 of the DCD. This was identified as Open Item OI-SRP9.1.3-SBPA-11 in the SER with open items.

In the RAI-SRP9.1.3-SBPA-11 response letter dated October 2, 2009, the applicant clarified that the high and low level alarms were not removed from the design; these alarms will be available to alert operators of SFP water level fluctuations. In the RAI response, the applicant proposed to change DCD Tier 2, Section 9.1.3.7 to clarify that the nonsafety-related instrumentation and alarms are available to alert operators. The staff verified that the High and Low level alarms were never intended to be safety-related alarms; therefore, the change in instrumentations does not impact the previous safety conclusion. Thus, Open Item OI-SRP-9.1.3-SBPA-11 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff noted that, in the AP1000 DCD Revision 17, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," the applicant lists each containment penetration and provides a summary of the containment isolation characteristics. In DCD Table 6.2.3-1 Sheet 2 of 4, the applicant identifies the containment isolation valves related to the SFP. Valve

SFS-PL-V067 is a pressure release valve located between SFS-PL-V034 and SFS-PL-V035. In DCD Tier 1 Table 2.2.1-1, the applicant also identifies the same pressure release valve but DCD Tier 2 Figure 9.1-6, "Spent Fuel Pool Cooling System Piping and Instrumentation Diagram," Sheet 1 of 2 does not show valve SFS-PL-V067. In RAI-SRP9.1.3-SBPA-12, the staff asked the applicant to update Figure 9.1-6, Sheet 1 to include valve SFS-PL-V067.

In its response dated August 25, 2009, the applicant agreed to revise DCD Figure 9.1-6 in the next DCD revision and illustrated the change in its response. The staff finds the change to be acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.3.2.7 Modification of the limiting site interface air temperatures

The applicant modified the wet bulb air temperature in AP1000 DCD, Section 9.1.3.1.3 and the discussion on exposure rates to personnel in Section 9.1.3.1.4. These changes are evaluated in Section 2.3.1, "Regional Climatology"; Section 12.2, "Ensuring that Occupational Radiation Doses Are as Low as Is Reasonably Achievable"; and Section 12.4, "Radiation Protection Design," of this report. The site temperature (wet and dry bulb) impacts the cooling tower performance, which affects the temperature of the component cooling water system (CCS). The SFS heat exchanger (HX) is cooled by the CCS, and a change in the CCS temperature affects the performance of the SFS.

Westinghouse Impact Document 36 (APP-GW-GLE-036), "Impact of a Revision to the Current Wet Bulb Temperature Identified in Table 5.0-1 (Tier 1) and Table 2-1 (Sheet 1 of 3) of the DCD (Revision 16)," Revision 0 of June 27, 2008, documents the impact of increasing the maximum safety wet bulb non-coincident temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F) and the maximum safety wet bulb coincident temperature from 26.7 °C (80 °F) to 30.1 °C (86.1 °F). The TR indicates that the SFP heat removal capability has been affected by this change. The thermal analysis results presented in the TR demonstrate that the SFS is capable of maintaining the SFP below 48.9 °C (120 °F) following a partial core fuel shuffle refueling, with the wet bulb temperature at the maximum safety limit (most limiting case). For a full core offload scenario, the thermal analysis demonstrated that the SFS is capable of maintaining the SFP at or below 48.9 °C (120 °F) based upon a service water heat sink at a maximum normal ambient wet bulb temperature at or below 60 °C (140 °F).

During the July 16, 2010, regulatory audit of the thermal calculation the staff discussed with the applicant the need to clarify DCD Tier 2 Section 9.1.3.1.3.1 "Partial Core." This section describes that the thermal analysis for the SFP cooling partial core scenario is based on a CCS supply temperature limited by the maximum normal ambient wet bulb temperature. The staff considered that this statement was not entirely correct, as while the reactor is at power, the CCS temperature limit is based on the maximum safety ambient wet bulb temperature.

In Open Item OI-SRP9.1.3-SBPA-13, Additional Question 3, the staff requested that the applicant clarify that DCD Section 9.1.3.1.3.1 properly represents the thermal analysis basis of the SFS. In a response letter dated August 20, 2010, the applicant presented the markups for modifying DCD Section 9.1.3.1.3.1 and Section 9.1.3.1.3.2 to clearly indicate when either the safety limit or the normal limit is credited. The staff found that the applicant's response properly addresses the staff concerns and clarified the DCD statement that could cause confusion. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.3.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of AP1000 DCD, Section 9.1.3, "Spent Fuel Pool Cooling System," the staff identified acceptance criteria based on the design meeting relevant requirements in 10 CFR Part 50, Appendix A, GDC 2; GDC 4; GDC 5; GDC 44, "Cooling Water"; GDC 45, "Inspection of Cooling Water Systems"; GDC 46, "Testing of Cooling Water Systems"; GDC 61; GDC 63; and in 10 CFR 20.1101(b), "Radiation protection programs," as it relates to radiation dose being kept as low as is reasonably achievable (ALARA). The staff found that the AP1000 SFS design was in compliance with these requirements, as referenced in NUREG-0800 Section 9.1.3 and determined that the design of the AP1000 SFS, as documented in AP1000 DCD, Revision 15, was acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 SFS as documented in AP1000 DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 9.1.3. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 SFS to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD. The staff concludes that the AP1000 SFS design continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criteria of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD. Therefore, the staff finds that the proposed changes to the AP1000 SFS are acceptable.

9.1.4 Light Load Handling System (Related to Refueling)

9.1.4.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.1.4, "Light Load Handling Systems" (LLHS) of the AP1000 DCD, Revision 15. In AP1000 DCD, Revision 17, the applicant has proposed to make changes to Section 9.1.4 of the DCD.

The applicant proposed 5 technical changes in the DCD supported by information presented in TR-106 and TR-121.

The staff reviewed the following changes to the AP1000 DCD, Section 9.1.4:

1. The applicant proposed to change the name, type and crane capacity of the new-fuel handling crane. In the previously approved DCD Revision 15, the new-fuel handling crane was called the fuel handling jib crane with a crane capacity of 907 kg (2000 lb). In DCD Revision 17, the applicant changed the name of the fuel handling jib crane to the new-fuel handling crane since the final crane specified may not be a jib crane. The new-fuel handling crane capacity was increased to lift a new fuel assembly, control rod assembly and handling tool [total weight of 919 kg (2,027 lb]. The basis for this change was documented in TR-106. In its June 26, 2008, response to RAI-SRP9.1.4-SBPB-01,

the applicant stated that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated.

The applicant identified this change in AP1000 DCD, Tier 2, Sections 9.1.1.1, 9.1.1.2, 9.1.1.2.1, and 9.1.1.3.

2. The applicant proposed to change the safety and seismic classification of the FHM and SFHT. In the previously approved DCD Revision 15, the FHM and SFHT were classified safety Class C, seismic Category I. The applicant changed the classification for both items to AP1000 Class D nonsafety-related, seismic Category II. The basis for this change is documented in TR-106.

   The applicant identified this change in DCD Tier 2, Sections 3.2.2.5, 3.2.2.6, and Table 3.2-3.

3. The applicant proposed to change the description of the FHM. In the previously approved DCD Revision 15, the FHM was described as “the same design as the refueling machine and includes the same safety features.” The applicant changed the description of the FHM to “the fuel handling machine has the same design functions as the refueling machine and includes the same safety features.” The basis for this change is documented in TR-106. This section was later changed in response to RAI-SRP-9.1.4-SBPB-03 to provide of the FHM safety features.

   The applicant identified this change in DCD Tier 2, Sections 9.1.4.3.3.

4. The applicant proposed to change Tier 1, Table 2.1.1-1, “ITAAC Acceptance Criteria for Design Commitment 5.” The revised acceptance criterion would state that “[t]he bottom of the dummy fuel assembly cannot be raised to within 24 ft, 6 in. [7.46 m] of the operating deck floor.” This change is documented in TR-121. This height restriction was later revised as reviewed in Section 9.1.4.2.4 below.

5. The applicant indicated that due to the radius of the FHM manipulator mast and the proximity to the SFP walls, approximately 25 percent of the SFP storage cells cannot be serviced by the mast crane. Also, there are instances where fuel inspection and/or fuel repair require the fuel to be moved from the SFP storage racks to the designated fuel inspection or fuel repair workstation. These non-normal fuel transfer operations are performed using the SFHT. The SFHT is a long handled tool, which latches onto the fuel assembly top nozzle via manually actuated grippers. Lifting of the SFHT and attached fuel assembly is performed using a hoist on the FHM. The applicant later changed the FHM configuration to eliminate the need for the mast crane and replace it with two hoists to handle fuel with the SFHT and new-fuel handling tool. This is reviewed in Section 9.1.4.2.5 below.

6. The applicant proposed to eliminate from the AP1000 design the mounting of the rod cluster control storage station from the reactor cavity wall and, therefore, its intended use. In DCD Section 9.1.4, the applicant has changed the title of paragraph K under the section heading Component Description from “Rod Cluster Control Storage Station” to “Not used.” In addition, the description of the rod cluster control storage station in paragraph K has been removed.

7. The applicant proposed to add the spent fuel assembly handling tool to the list of tools itemized under the section heading Fuel Handling Tools and Equipment. In the DCD Section 9.1.4, the applicant added to this section paragraph C with the title “Spent Fuel Assembly Handling Tool.” The new paragraph describes some operational aspects of the tool and preoperational testing.

   The applicant identified this change in DCD Tier 2, Section 9.1.4.3.4.

8. In DCD Revision 17, the applicant proposed to change the refueling water and reactor coolant nominal boron concentration. The applicant completed the “Spent Fuel Storage Racks Criticality Analysis,” for the new SFP racks and documented its results in TR-65. The applicant proposed the changes to boron concentration in DCD Section 9.1.4 to be consistent with the analyses presented in TR-65. This change is evaluated in Section 9.1.2.2.4 of this report.

9. In DCD Revision 17, Sections 9.1.4.2.3 and 9.1.4.3.7, the applicant changed the distance between the top of the active fuel to the surface of the spent fuel water. In Section 9.1.4.3.7, the applicant proposed to change the dose rate at the surface of the water during spent fuel transfer from 20 millirem/hour or less to an exposure rate for an operator to 2.5 millirem per hour or less. These changes are reviewed in Section 12.3 of this report.

9.1.4.2 Evaluation

The staff reviewed all changes to the LLHS in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 9.1.4, “Light Load Handling System (Related to Refueling).” The regulatory basis for Section 9.1.4 of the AP1000 DCD is documented in NUREG-1793, which states that staff acceptance of the design, is contingent on compliance with the following requirements:

- GDC 2, as it relates to the ability of structures, systems, and components (SSCs) to withstand the effects of earthquakes

- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions

- GDC 61, as it relates to a radioactivity release resulting from fuel damage and the avoidance of excessive personnel radiation exposure

- GDC 62, as it relates to criticality prevention

The specific criteria that apply to the proposed DCD changes are; 10 CFR 52.63(a)(1)(vi), which concerns substantially increasing overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii) which concerns contribution to the increased standardization of the certification information in the AP1000 DCD.

9.1.4.2.1 Name Change and Crane Capacity Change for New-Fuel Handling Crane

The applicant proposed to change the name, type and crane capacity of the new-fuel handling crane. In Revision 17 to the DCD, the applicant deleted the classification of new-fuel handling crane as a jib crane. Based on TR-106, the applicant made this change because the final crane specified may not be a jib crane. Accordingly, the applicant changed the name of this crane to the new-fuel handling crane (from the fuel handling jib crane). The applicant also changed the capacity of the new-fuel handling crane from 907 kg (2000 lb) to 919 kg (2027 lb), which is the total combined weight of a new fuel assembly, control rod assembly and handling tool. Subsequently, the applicant stated in a letter dated June 26, 2008, that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated. This issue is discussed below.

In accordance with NUREG-0800 Section 9.1.4, the LLHS needs to meet the requirements of GDC 2, GDC 5, GDC 61 and GDC 62. The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 61 and GDC 62 are based on meeting the guidelines of ANSI/ANS 57.1-1992, "Design Requirements for Light Water Reactor Fuel Handling Systems." The staff finds that by changing the name of the new-fuel handling crane, removing its designation as a jib crane, and increasing its capacity, the applicant will continue to comply with the requirements of GDCs 5, 61 and 62.

The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 2 are based on compliance with meeting the Regulatory Positions C.1 and C.2 of RG 1.29. The new-fuel handling crane, which handles new fuel and loads the new fuel into the SFP, was designed to be a seismic Category II component. Regulatory Position C.2 of RG 1.29 and Section 3.2.1.1.2 of DCD Revision 17, describe the guidance for seismic Category II SSCs. These state, in part, that seismic Category II SSC should be designed to preclude their structural failure during a safe shutdown earthquake (SSE) or interaction with seismic Category I items that could degrade the functioning of safety-related SSCs to an unacceptable level. In order to be seismic Category II, DCD Sections 9.1.1.2.1.D and 9.1.2.2.1.E state that the new-fuel handling crane will neither fall into the new fuel storage pit nor collapse into the SFP during a seismic event. Although the new-fuel handling crane will neither fall nor collapse during a seismic event as stated above, DCD Sections 9.1.1.2.1.D and 9.1.2.2.1.E did not state that the new fueling handling crane will continue to hold its maximum load (not drop the load) during the seismic event. Since a load drop could degrade the functioning of safety-related SSCs to an unacceptable level, the staff asked the applicant in RAI-SRP9.1.4-SBPB-01 to explain how this crane will meet seismic Category II criteria considering the maximum load carried by the crane.

The applicant responded in a letter dated June 26, 2008, indicating that the function of moving new fuel will be transferred to the FHM and that the new-fuel handling crane will be eliminated. The applicant stated that the FHM will be changed from a sigma style crane to a bridge style crane. The applicant stated that the FHM will remain a seismic Category II component and will not drop a fuel element under SSE conditions. The staff found the applicant's response satisfactory because the function of the new-fuel handling crane will be performed by the FHM and the FHM will continue to be a seismic Category II component in order to not drop its load during an SSE. Thus, the applicant will need to comply with GDC 2 when the new-fuel handling crane is eliminated and replaced by the FHM.

The applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.4-SBPB-01 in a letter dated April 3, 2009. The applicant's revised response states that the words, "The fuel handling machine is designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake," will be inserted into DCD Tier 2, Section 9.1.1.2.1.D, "New Fuel Rack Design," and DCD Tier 2, Section 9.1.2.2.1.E, "Spent Fuel Rack Design." In the response, the applicant provided markups of DCD Tier 2, Sections 9.1.1.2.1 D and 9.1.2.2.1 E, which showed how the additional text will be added in the next DCD revision. However, in Revision 1 to the RAI response the applicant also made the statement that these changes support the statement in DCD Tier 1, Section 2.1.1, "Fuel Handling and Refueling System," Item 6, "The RM (refueling machine) and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake." While the Revision 1 response addressed the FHM seismic capabilities, the RM seismic capabilities were not addressed even though the RAI response implied that they would be addressed.

In a letter dated September 4, 2009, the applicant provided Revision 2 to the response to RAI-SRP9.1.4-SBPB-01. The applicant's revised response stated that the words, "the refueling machine is designed to maintain its load carrying and structural integrity functions during a safe shutdown earthquake" will be included in Tier 2 of the DCD. The staff finds this acceptable because the DCD will address both RM and FHM ability to hold their load during an SSE, and the DCD will provide Tier 2 seismic information to support Tier 1 content. Therefore, RAI-SRP9.1.4-SBPB-01 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.4.2.2 Non Safety-related Classification for AP1000 Fuel Handling Equipment:

The applicant proposed to change the safety and seismic classification of the FHM and SFHT. As documented in TR-106, the applicant reclassified the FHM and the SFHT as nonsafety-related, seismic Category II. As stated in Section 9.1.4.1.1 of this report, NUREG-0800 Section 9.1.4 states that the LLHS needs to meet the requirements of GDC 2, GDC 5, GDC 61 and GDC 62. The staff determined that this change does not affect the LLHS meeting the requirements of GDC 5 because this change will not cause the sharing of equipment important to safety between nuclear power units.

The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 61 and GDC 62 are based on meeting the guidelines of ANSI/ANS 57.1-1992. The staff finds that the nonsafety classification of the FHM and the SFHT is consistent with ANSI/ANS 57.1-1992. Therefore, the staff finds the reclassification of FHM and the SFHT as nonsafety-related to be acceptable.

The NUREG-0800 Section 9.1.4 acceptance criteria for meeting the requirements of GDC 2 are based on compliance with meeting the Regulatory Positions C.1 and C.2 of RG 1.29. Since these SSCs have been accepted by the staff as nonsafety-related, Regulatory Position C.2 of RG 1.29 is applicable in meeting the requirements of GDC 2. Regulatory Position C.2 states that seismic Category II SSCs need to be designed to preclude their failure, which could reduce the function of any seismic Category I SSC during an SSE to an unacceptable safety level.

The staff asked the applicant in RAI-SRP9.1.4-SBPB-02, to verify that the FHM and the SFHT will continue to hold their design load during an SSE. The applicant responded in a letter dated June 26, 2008, that the FHM will be designed to maintain its structural integrity and load carrying ability during an SSE. The staff found the applicant's response satisfactory because it complies with Regulatory Position C.2 of RG 1.29 and thus, the requirements of GDC 2 are met. Although the AP1000 DCD, Revision 16 stated that the FHM maintains its structural integrity during an SSE, the DCD did not state that the FHM maintains its load carrying capability during an SSE.

The applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.4-SBPB-02, dated April 3, 2009. The applicant's revised response referred to the changes made in its Revision 1 response to RAI-SRP9.1.4-SBPB-01. As stated above, while the Revision 1 response to RAI-SRP9.1.4-SBPB-01 addressed the FHM seismic capabilities, the RM seismic capabilities were not addressed as the RAI response implied. The applicant's September 4, 2009, revised response to RAI-SRP9.1.4-SBPB-01 included information related to RAI-SRP9.1.4-SBPB-02 addressing the RM seismic capability. The staff finds this response acceptable because it ensures the RM and FHM seismic capability to hold load during an SSE.

However, in DCD Revision 16, Table 3.2-3, the applicant designated dual seismic classification to the FHM. This was the only SSC in Table 3.2-3 that had a dual seismic classification of "II/NS," and no reason for the dual seismic classification was given by the applicant. Therefore, the staff requested in RAI-SRP9.1.4-SBPB-02 that the applicant provide justification for the dual seismic classification. The applicant responded in a letter dated June 26, 2008, that the FHM was a seismic Category II component and that Table 3.2-3 would be updated in Revision 17 of the DCD. The applicant provided a DCD markup page of Table 3.2-3. The staff has confirmed that Revision 17 of the DCD Tier 2, Table 3.2-3 was revised as committed in the RAI response. Furthermore, the staff finds that the issue concerning the seismic classification of the FHM has been adequately addressed and resolved by the applicant, because the FHM complies with Regulatory Position C.2 of RG 1.29 and thereby the requirements of GDC 2. Therefore, the staff's concern described in RAI-SRP9.1.4-SBPB-02 is resolved.

With the designation of the FHM and RM as seismic Category II, the staff finds that the LLHS will continue to meet the requirements of GDC 2 as it relates to the ability of SSCs to withstand the effects of earthquakes, GDC 61 as it relates to adequate safety from radioactivity resulting from fuel damage, and GDC 62 as it relates to prevention of criticality.

9.1.4.2.3 Fuel Handling Machine Generic Description

The applicant proposed to change the safety evaluation description of the FHM. In accordance with TR-106, the applicant revised the description of the FHM. In DCD Revision 16, Section 9.1.4.3.3, the applicant stated that the FHM "has the same design functions as the RM and includes the same safety features." However, DCD Revision 16, Sections 9.1.4.2.4 and 9.1.4.2.2.3 stated that the RM services the core—including the function to latch and unlatch control rods. No such function was attributed to the FHM. Additionally, DCD Revision 16, Section 9.1.4.2.3 stated that the FHM is used to load spent fuel into the shipping casks. No such function was attributed to the RM. Additionally, the RM operated exclusively in containment while the FHM operated exclusively in the fuel handling area. Therefore, the staff requested, in RAI-SRP9.1.4-SBPB-03, that the applicant explain how the FHM has the same design functions as the RM.

The applicant responded in a letter dated June 26, 2008, that the FHM design will be changed to a bridge/gantry style machine with two 2-ton overhead hoists. The applicant then described the safety interlocks, bridge hold-down devices, hoist braking system, and the fuel assembly support system for the FHM. The staff addresses the adequacy of the applicant's description of the FHM in Section 9.1.4.2.5 of this report.

9.1.4.2.4 Tier 1, Table 2.1.1-1, ITAAC Acceptance Criteria for Design Commitment 5

In the previously approved AP1000 DCD Revision 15, Tier 1, Table 2.1.1-1, ITAAC Acceptance Criteria for Design Commitment 5 stated that "[t]he bottom of the dummy fuel assembly cannot be raised to within 26 ft, 1 in. [7.9 m] of the operating deck floor." In DCD Revision 17, Tier 1, Table 2.1.1-1, the applicant proposed to change the lift height to 7.46 m (24 ft, 6 in). The bases for these changes are included in TR-121, and these changes are reviewed by the staff in Section 9.1.3 and Section 12.3 of this report. Some of the potential effects on DCD Section 9.1.4 are summarized in the applicant's response to RAI-SRP9.1.4-SBPB-04, Revision 1. The staff determined that these changes made to DCD Tier 2, Section 9.1.4 and Tier 1, Table 2.1.1-1 are conforming changes to the changes made in DCD Section 9.1.3 and do not impact the staff's safety evaluation of DCD Section 9.1.4. In addition, the staff's evaluation of the justification for the minimum shielding change is discussed in the applicant response to RAI-SRP12.3-CHPB-02 and is reviewed and documented in Section 12.3 of this report. The new lift height limit of 7.46 m (24 ft, 6 in) continues to demonstrate that the FHM will not raise a fuel assembly above the minimum required depth of water shielding.

9.1.4.2.5 Moving Spent Fuel With the SFHT and Auxiliary Hoist of the FHM

NUREG-0800 Section 9.1.4, "Light Load Handling System (Related to Refueling)," invokes GDC 61 for avoidance of excessive personnel radiation exposure. NUREG-0800 Section 9.1.4 acceptance criteria for meeting the relevant aspects of GDC 61 are based in part on the guidelines of ANSI/ANS 57.1-1992. Section 6.1.1 of this standard states, "Mechanical or electrical safety devices shall be designed into the system to prevent damage to fuel units and conditions that pose a radiation hazard or an unintentional radiation exposure risk to personnel." Section 6.4.1.2 of this standard recommended testing of these safety devices. In RAI-SRP9.1.4-SBPB-04, the staff asked the applicant to explain how they meet ANSI/ANS 57.1-1992 for the auxiliary hoist of the FHM, and the staff asked what ITAAC will test the functionality of the aforementioned safety devices.

The applicant responded to RAI-SRP9.1.4-SBPB-04 in a letter dated June 26, 2008, that the mast and auxiliary hoist was eliminated from the FHM and replaced with two overhead/trolley hoists. Since the applicant eliminated the mast and replaced it with an overhead hoist, the applicant needed to address the effects upon safe movement of fuel without the stability and position accuracy of a mast, in not causing mechanical damage to the new and spent fuel assemblies during movements of fuel assemblies. Movements of fuel within the refueling area include loading spent fuel into the spent fuel racks and moving spent and new fuel between the SFP and the fuel transfer canal.

The applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.4-SBPB-04 in a letter dated May 20, 2009. The applicant's revised response stated that use of an overhead hoist design with an SFHT is common industry practice and has been accepted in previous designs. The applicant further stated: (1) the FHM is unable to raise the fuel to an unsafe radiological height in the SFP at any time during transport, storage, and installation activities; (2) features of the lifting devices' control circuitry and procedural operation prevent inadvertent unsafe movement of the fuel; (3) fuel transit speed, as operating limits or obstructions are approached, is automatically reduced to prevent overtravel or excessive sway of a fuel assembly. The staff finds this method of moving spent fuel with the SFHT acceptable since this type of design is consistent with common industry practice and is currently used in operating plants.

The applicant stated in a letter dated June 26, 2008, that both FHM hoists will be equipped with the safety devices identified in ANSI/ANS 57.1-1992, Section 6.3.1 with the exception of (9) Grapple Release as this feature is the manual operation of the fuel handling tool. A staff review of the applicant's proposed changes to the DCD, as described in its response to RAI-SRP9.1.4-SBPB-04, found that the down-position (interlock), end-travel (hardstop), up-limit (hardstop), translation inhibit (interlock) were not described. Furthermore, the applicant was not clear in its response whether the bridge travel (interlock) and trolley travel (interlock) are part of the design of the FHM. The applicant needed to clarify in the DCD that the FHM hoists have the safety devices identified in ANSI/ANS 57.1-1992, Section 6.3.1.

In its May 20, 2009 revised response to RAI-SRP9.1.4-SBPB-04, the applicant also proposed to include a description that clarifies the safety interlocks for the FHM in Section 9.1.4.3.3. Additional content was proposed in the RAI- SRP9.1.4-SBPB-03 response dated June 4, 2009 to incorporate a description of FHM bridge hold-down devices, hoist braking system, and the fuel assembly support system into Section 9.1.4.3.3. The applicant also stated that portions of the DCD "Light Load Handling System (Related to Refueling)" text will be clarified to indicate that the safety features listed include the safety requirements listed in ANSI/ANS 57.1. The clarifying text, which states that the safety features listed include the safety requirements listed in ANSI/ANS 57.1, will be added to DCD Sections 9.1.4.3.1, "Refueling Machine"; 9.1.4.3.2, "Fuel Transfer System"; and 9.1.4.3.3, "Fuel Handling Machine." In Section 9.1.4.3.3, there is one exception to ANSI/ANS 57.1 for the grapple release, as this feature is the manual operation of a fuel handling tool. In its response, the applicant provided markups of DCD Tier 2 Sections 9.1.4.3.1, 9.1.4.3.2 and 9.1.4.3.3 to be added in the next revision to the DCD. The staff finds this response acceptable since the safety requirements listed in ANSI/ANS 57.1 will be applied to the RM, fuel transfer system, and FHM; this meets GDC 61 and GDC 62. The staff's concern described in RAI-SRP9.1.4-SBPB-04 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

To address the part of the staff's RAI-SRP9.1.4-SBPB-04 request to identify the ITAAC that will test the functionality of the auxiliary hoist of the FHM, the applicant proposed, in Revision 17 of DCD Tier 1, changes to Section 2.1.1, "Fuel Handling and Refueling System," paragraphs 4, 5 and Table 2.1.1-1 (ITAAC) for associated Design Commitments 4 and 5. Based on its review, the staff finds that the changes to the above ITAAC are acceptable since they will adequately verify that the FHM/SFHT gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers. ITAAC Design Commitment 5 will also verify that the FHM hoists are limited such that the minimum required depth of water is maintained.

Single failure and non-single-failure proof hoists

In the June 26, 2008, response to RAI-SRP9.1.4-SBPB-04, the applicant stated that the FHM design will be changed to a bridge style machine with two 2-ton overhead hoists, one of which is single-failure proof. It was stated that a single-failure proof hoist and the new fuel handling tool will be used to handle new fuel and a nonsingle-failure proof hoist and the SFHT will be used to handle spent fuel. The applicant initially stated that the single-failure proof hoist may also handle spent fuel, but it would not have access to all spent fuel handling/storage locations. In a March 18, 2009, public meeting between the staff and the applicant, the use of the FHM single-failure proof hoist and nonsingle-failure proof hoist was discussed in detail. The applicant stated that the new FHM will handle new fuel and spent fuel.

In the June 26, 2008 response to RAI-SRP9.1.4-SBPB-03, the applicant also stated, “The fuel handling machine is restricted to raising a fuel assembly to a height at which the water provides a safe radiation shield,” and in response to RAI-SRP9.1.4-SBPB-04, the applicant stated that “each FHM hoist will have a mechanical limit based on maximum hoist up travel and spent fuel handling tool length.” Since the new FHM will be moving both new fuel and spent fuel, and new fuel is handled above deck level when it is transferred to the new fuel racks and transferred from the new fuel storage vault into the SFP, the applicant did not state in the DCD how the same cranes that are restricted in hoist uptravel can handle new fuel above deck level. Use of the FHM hoist for new fuel also apparently conflicted with the revised Tier 1 ITAAC Table 2.1.1-1 Item 5, which stated, “FHM hoists are limited such that the minimum required depth of water shielding is maintained.”

The applicant provided Revision 1 to its response to RAI-SRP9.1.4-SBPB-04 in a letter dated May 20, 2009 and Revision 1 to its response to RAI-SRP9.1.4-SBPB-03 in a letter dated June 4, 2009. Both of the applicant’s revised RAI responses include the same additional paragraph, which defined some restrictions to the use of the nonsingle-failure proof hoist and the single-failure proof hoist of the FHM for handling new fuel and other loads throughout the fuel handling area. The single-failure proof hoist in conjunction with the SFHT is not capable of raising spent fuel to a height that clears the spent fuel racks, fuel transfer system fuel basket, spent fuel shipping cask, or the new fuel elevator. The staff found that the applicant’s Revision 1 responses to RAI-SRP9.1.4-SBPB-03 and RAI-SRP9.1.4-SBPB-04 did not adequately address how the single-failure proof crane of the FHM with hoist uptravel restrictions can handle new fuel above the deck level.

The applicant’s Revision 2 to its RAI-SRP9.1.4-SBPB-03 response was issued in a letter dated October 15, 2009. The letter provides a clarifying description of the fuel movement process using the FHM. The design for the movement of spent fuel utilizes the nonsingle-failure proof hoist and its associated SFHT. The new fuel handling tool is used with the single-failure proof hoist to move new fuel. The SFHT and the new fuel handling tool are manually operated tools and differ in length by approximately 9.14 m (30 ft). The single-failure proof hoist does not have the lift height to raise a spent fuel assembly clear of the spent fuel racks, fuel transfer system fuel basket, spent fuel shipping cask, or the new fuel elevator when using the SFHT. When spent fuel is stored in the spent fuel racks, or other interim storage locations, spent fuel movement with either hoist would be physically impossible using the new fuel handling tool, as the operating handle of the tool would be submerged in approximately 6.09 m (20 ft) of water.

Therefore, a description of the fuel movement (new and spent) process for both FHM hoists using their handling tools, and a discussion of their interlocks, needed to be included in the DCD. This was identified as Open Item OI-SRP9.1.4-SBPA-03.

To address Open Item OI-SRP9.1.4-SBPA-03, the applicant provided a response dated March 31, 2010. The response proposed to revise DCD Section 9.1.4.2.4 to incorporate the statement provided in RAI-SRP9.1.4-SBPB-03 and RAI-SRP9.1.4-SBPB-04 regarding the restrictions for use of single and nonsingle-failure proof hoists in the DCD. However, when further requested to clearly define the usage of the single and nonsingle-failure proof hoists, the applicant submitted a revised response to Open Item OI-SRP9.1.4-SBPA-03 dated May 19, 2010. This revised response proposed to revise DCD Section 9.1.4.2.4 to define the use of hoists above deck. This response also described the availability of a selector switch to choose the correct hoist and integrated controls to avoid inadvertent use of the incorrect hoist. The response also defined the intended use of each hoist as follows:

The single-failure proof hoist will be used for;

- primarily handling new fuel
- the movement of loads <1814 kg (4000 lb) in the fuel handling area of the auxiliary building
- a redundant hoist over the SFP for the handling of control components

The nonsingle-failure proof crane will be used for;

- handling fuel and control components in the SFP
- the hoist shall be restricted from handling a load above the operating floor within 4.6 m (15 ft) of the SFP

The proposed additions to Section 9.1.4.2.4 clarify that the nonsingle-failure proof hoist is primarily used for submerged handling activities in the SFP. There are areas in the fuel handling area of the auxiliary building that the single-failure proof hoist is not capable of accessing due to travel limitations. Therefore, the response stated that it is necessary for the nonsingle-failure proof hoist to be used in areas other than the SFP. As a safety precaution, the applicant indicated that the nonsingle-failure proof hoist will be restricted from handling a load above the operating floor within 4.6 m (15 ft) of the SFP. The applicant further stated that the single-failure proof hoist will be capable of handling loads in the new fuel handling area and the spent fuel handling area with operator warnings associated with the handling of spent fuel and proposed corresponding changes to DCD.

Although the proposed DCD changes addressed restriction on the nonsingle-failure proof hoist travelling over SFP, it was unclear what provisions are provided to prevent the single-failure proof hoist from handling new fuel over SFP with the new fuel handling tool. For additional clarification of new fuel movement, the applicant provided a revised response to Open Item OI-SRP9.1.4-SBPA-03 dated July 9, 2010, which further defined the restrictions of new fuel movement above deck while using the new fuel handling tool. The open item response proposed a revision to the DCD to indicate that the nonsingle-failure proof hoist is restricted from handling new fuel above the operating floor. The applicant also clarified that the new fuel elevator fuel carrier is located in the tool storage area of the SFP. The open item response revised DCD Figure 9.1-4 to indicate the location of the new fuel elevator and defined the safety interlocks to prevent the transporting of new fuel above the operating floor over the spent fuel racks by the single-failure proof hoist in Section 9.1.4.2.4.

Based on the above, the staff finds that there is reasonable assurance that the use of the single and nonsingle-failure proof hoist configuration on the FHM will minimize the potential for damage to fuel, fuel assemblies, and to storage or transport containers. The restrictions on fuel movement and safety interlocks provide safe movement of new and spent fuel in the auxiliary building using these hoists. Therefore, Open Item OI-SRP9.1.4-SBPA-03 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

To address the section part of the staff’s RAI-SRP9.1.4-SBPB-04 to identify the ITAAC that will test the functionality of the auxiliary hoist of the FHM, the applicant proposed, in Revision 17, Tier 1, changes to DCD Tier 1, Section 2.1.1, “Fuel Handling and Refueling System” paragraphs 4 and 5 and DCD Tier 1 Table 2.1.1-1 (ITAAC) for associated Design Commitments 4 and 5. Based on its review, the staff finds that the changes to the above ITAAC are acceptable.

Table 9.1-1 in Section 9.1.4.2.8 of this report provides a summary of the fuel handling operations of the FHM discussed above.

9.1.4.2.6 Elimination of Rod Cluster Control Storage Station from Reactor Cavity Wall

In DCD Tier 2, Revision 17, the applicant proposed to eliminate from the AP1000 design the mounting of the rod cluster control storage station from the reactor cavity wall and, therefore, its intended use. The applicant proposed that it will instead perform rod cluster control assembly inspections in the auxiliary building. As stated in Section 9.1.4.1.1 of this report, NUREG-0800 Section 9.1.4 states that the LLHS needs to meet the requirements of GDC 2, GDC 5, GDC 61 and GDC 62. The staff determined that this change of eliminating the rod cluster control storage station from the reactor cavity wall will have no effect on the remaining LLHS components from meeting the applicable GDC.

9.1.4.2.7 Addition of Spent Fuel Assembly Handling Tool under Fuel Handling Tools and Equipment Section

In DCD Tier 2, Revision 17, the applicant proposed to add the spent fuel assembly handling tool to the list of tools itemized under the Section 9.1.4.3.4. This change is associated with the comprehensive changes made to Section 9.1.4 by the applicant due to the elimination of the new-fuel handling crane as discussed in the applicant's responses to RAI-SRP9.1.4-SBPB-01 through RAI-SRP9.1.4-SBPB-04. The staff determined that the addition of the spent fuel assembly handling tool to the list of tools itemized under the section heading fuel handling tools and equipment is acceptable, because it represents additional changes needed to Section 9.1.4, which were not previously presented in the responses to RAI-SRP9.1.4-SBPB-01 through RAI-SRP9.1.4-SBPB-04.

9.1.4.2.8 Fuel Handling Summary Table

Table 9.1-14. Fuel Handling Machine (FHM) Hoist Operations

Fuel Handling Scenario	Fuel Handling Machine (FHM) Hoist Operations			
	Non-Single-Failure-Proof Hoist		Single-Failure-Proof Hoist	
	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool
Moves new fuel (NF) above the fuel handling area (FHA) operating floor/deck over spent fuel (SF) racks.	No	No	No	No
Moves NF above the FHA operating floor/deck.	No	No	**Yes**	No

Table 9.1-14. Fuel Handling Machine (FHM) Hoist Operations

Fuel Handling Scenario	Fuel Handling Machine (FHM) Hoist Operations			
	Non-Single-Failure-Proof Hoist		Single-Failure-Proof Hoist	
	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool	New Fuel Assembly Handling Tool	Spent Fuel Assembly Handling Tool
Moves NF over the NF racks. (Same as Scenario 2)	No	No	**Yes**	No
Moves NF over the NF Elevator (elevator up).	No	No	**Yes**	No
Moves NF over the NF Elevator (elevator down).	No	**Yes**	No	No
Moves NF within the SFP over the SF racks.	No	**Yes**	No	No
Moves SF within the SFP over the SF racks.	No	**Yes**	No	No

9.1.4.3 Conclusion

In NUREG-1793 and Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 LLHS as documented in DCD, Revision 17 against the relevant acceptance criteria as listed above and in NUREG-0800 Section 9.1.4. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 LLHS to meet the applicable acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. The staff concludes that the AP1000 LLHS design continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criteria of: 10 CFR 52.63(a)(1)(vi), on the basis that they substantially increase overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD. Therefore, the staff finds that the proposed changes to the AP1000 LLHS are acceptable.

9.1.5 Overhead Heavy Load Handling Systems

9.1.5.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.1.5, "Overhead Heavy Load Handling Systems" (OHLHS) of the AP1000 DCD, Revision 15. In AP1000 DCD, Revision 17, the applicant has proposed to make changes to Section 9.1.5 of the DCD.

The applicant proposed changes in the DCD, which are supported by information presented in TR-106.

The staff reviewed the proposed changes to Section 9.1.5 of the AP1000 DCD, Revision 15.

1. The applicant proposed to rename the cask handling crane, upgrade its seismic classification, and make the cask handling crane single-failure proof. In the previously approved AP1000 DCD Revision 15, the applicant named this crane the spent fuel shipping cask crane. In DCD Revision 17, the applicant has changed the name of this crane to the cask handling crane, and changed the design basis to single-failure proof and seismic Category I.

   The applicant has also proposed to classify the maintenance hatch hoist as single-failure proof. In the previously approved AP1000 DCD Revision 15, this hoist was seismic Category I, but not single-failure proof. In DCD Revision 17, the applicant has added single-failure proof design criteria for the maintenance hatch hoist.

   In DCD Revision 17, the applicant also added two new safety-related functions for the mechanical handling system (MHS), those being the prevention of the uncontrolled lowering of a heavy load by both the cask handling crane and the maintenance hatch hoist. The applicant also added the cask handling crane and the maintenance hatch hoist to Tables 2.3.5-1 and 2.3.5-3 in Tier 1 to add these components to the tables that list seismic Category I equipment and component locations for the MHS. The applicant also added the cask handling crane and the maintenance hatch hoist to ITAAC in Table 2.3.5-2 of Tier 1.

   The bases for these changes are documented in the applicant's TR-106.

   The applicant identified these changes in AP1000 DCD Revision 17, Tier 1, Section 2.3.5, "Mechanical Handling System," and Tables 2.3.5-1, 2.3.5-2, and 2.3.5-3; Tier 2, Sections 9.1.5.1.1, "Safety Design Basis," 9.1.5.2, "System Description," and 9.1.5.3, "Safety Evaluation," Table 9.1-5, "Nuclear Island Heavy Load Handling Systems," and Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment."

2. The applicant proposed to modify the codes and standards applicable to the polar crane, cask handling crane and other overhead cranes and hoists. In the previously approved AP1000 DCD Revision 15, overhead cranes were designed to ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)." Other cranes and hoists that handle heavy loads were designed according to the applicable ANSI standard. AP1000 cranes were designed to ASME NOG-1 and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The spent fuel shipping cask crane was designed to ASME NOG-1 for a Type III crane and

ANSI/ANS-57.1 and ANSI/ANS-57.2, "Design Requirements Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." In DCD Revision 17, the applicant has changed the polar crane and cask handling crane to be designed according to NUREG-0554 supplemented by ASME NOG-1 for a Type I single-failure proof crane. Other cranes and hoists which handle heavy loads will be designed according to ASME NOG-1 and to the applicable ANSI standard. In addition, the design of AP1000 cranes will comply with the guidance in NUREG-0612. The bases for all these changes are documented in TR-106.

The applicant identified these changes in DCD Revision 17, Tier 2, Sections 9.1.5.1.2, "Codes and Standards," and 9.1.5.2.1.2, "Component Descriptions," and Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment."

3. The applicant proposed to change one method by which the movements of the bridge, trolley, main, and auxiliary hoists of the polar crane can be controlled. In DCD Revision 15, the polar crane was controlled from either the operator's cab or from a pendant suspended from the crane. In DCD Revision 17, the applicant has changed control from a pendant to remote control. The remote control of the crane will have the same control functions, push button functions, and interlocks as the previous pendant design. The basis for this change is documented in TR-106.

   The applicant has identified this change in DCD Revision 17, Tier 2, Section 9.1.5.2.1.1, "System Operation."

4. The applicant proposed a new description for the cask handling crane. In the previously approved DCD Revision 15, the applicant did not describe the cask handling crane, formerly called the spent fuel shipping cask crane. In DCD Revision 17, with the upgrade of the crane to single-failure proof and seismic Category I, the applicant added the following sections and titles to describe the cask handling crane and its operation:

   9.1.5.2.2 Cask Handling Crane General Description
   9.1.5.2.2.1 System Operation
   9.1.5.2.2.2 Component Descriptions.
   9.1.5.2.2.3 Instrumentation Applications

   The bases for these changes are documented in TR-106.

5. The applicant proposed to change the maximum load rating of the containment polar crane. In the previously approved DCD Revision 15, the maximum load rating of the containment polar crane was 249.48 metric tons (275 tons). In DCD Revision 17, the applicant proposed to change the maximum load rating for the containment polar crane to 272.16 metric tons (300 tons). The basis for this change is documented in TR-106.

   The applicant identified this change in DCD Revision 17 in Table 9.1-5, "Nuclear Island Heavy Load Handling Systems."

6. The applicant proposed to change the tabular information in Table 9.1.5-1 for the cask handling crane (previously spent fuel shipping cask crane in DCD Revision 15). In the previously approved DCD Revision 15, the table was for a nonsingle-failure proof spent fuel cask crane and listed exceptions and clarifications to ANSI/ANS 57.1 and ANSI/ANS 57.2. In DCD Revision 16, the applicant removes the exceptions table and

replaces it with design information for the bridge, trolley, main hoist and auxiliary hoist of the cask handling crane.

The applicant identified this change in DCD Revision 17 in Table 9.1.5-1, "Cask Handling Crane Component Data."

7. The applicant proposed to specify the main hook of the polar crane to handle the RCP pump/motor shell. In the previously approved DCD Revision 15, the auxiliary hook of the polar crane was used to handle the RCP pump/motor shell. In DCD Revision 17, the applicant proposed to change Table 9.1.5-2, "Special Lifting Devices Used for the Handling of Critical Loads," to designate the main hook of the polar crane to be used with the RCP special lifting device to lift the RCP pump/motor shell. The basis for this change is documented in TR-106.

   The applicant identified this change in Table 9.1.5-2, "Special Lifting Devices Used for the Handling of Critical Loads."

8. The applicant proposed to change the approximate capacity of the polar crane auxiliary hoist from 68.04 metric tons (75 tons) to 22.68 metric tons (25 tons). In the previously approved DCD Revision 15 in Table 9.1.5-3, "Polar Crane Component Data," 68.04 metric tons (75 tons) was listed as the approximate capacity of the auxiliary hoist of the polar crane. In DCD Revision 17, the applicant proposed to change the approximate capacity of the polar crane auxiliary hoist to 22.68 metric tons (25 tons). The basis for this change is documented in TR-106.

   The applicant identified this change in Table 9.1.5-3, "Polar Crane Component Data."

9.1.5.2 Evaluation

The staff reviewed all changes to the OHLHS in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 9.1.5, "Overhead Heavy Load Handling Systems." The staff did not re-review descriptions and evaluations of Section 9.1.5 of DCD Revision 15 that were previously approved and are not affected by the new changes. The regulatory basis for Section 9.1.5 of the DCD is documented in NUREG-1793.

The specific criteria that applies to the proposed DCD changes are; 10 CFR 52.63(a)(1)(vi), which concerns substantially increasing overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.1.5.2.1 Cask Handling Crane and Containment Maintenance Hatch Hoist Design Basis Changes (Single-failure Proof and Seismic Category I)

TR-106 states that the AP1000 cask handling crane design was neither seismically qualified, nor single-failure proof to protect against dropping a cask. Since a cask drop could cause significant plant damage, a design change was initiated to upgrade the cask handling crane to single-failure proof and seismic Category I. This change was made to reduce the possibility of a serious plant event. With this design change the crane name was changed from the spent fuel shipping cask crane to the cask handling crane. The cask handling crane will thus be designed to support a critical load during and after a safe shutdown earthquake and is classified as seismic Category I.

TR-106 further states that the maintenance hatch hoist [9.07 metric tons (10 ton capacity)] is not specified in the DCD as single-failure proof. For personnel and equipment safety reasons, the maintenance hatch hoist design has also been changed to be single-failure proof. The equipment hatch hoist which was already designated a single-failure proof crane had its name changed to containment equipment hatch hoist to be specific in its plant location.

On the basis that upgrading of the cask handling crane and maintenance hatch hoist to single-failure proof and upgrading the cask handling crane to seismic Category I will result in greater safety during the handling of critical loads within the nuclear island, the staff finds the design changes acceptable.

However, the applicant, in DCD Tier 1 Section 2.3.5, “Mechanical Handling System,” did not list single-failure proof as certified design information with ITAAC for the polar crane, the cask handling crane, the containment equipment hatch hoist or the containment maintenance hatch hoist. One design criterion for Tier 1 information is that it should include features and functions that could have a significant effect on the safety of a nuclear plant or are important in preventing or mitigating severe accidents. A drop of the reactor vessel head or a spent fuel cask could affect plant safety. Without heavy load drop analyses that proves the safety of a load drop, single-failure proof design criteria for the associated crane/hoist is necessary for plant safety and should be part of the Tier 1 safety significant design criteria for the polar crane and the cask handling crane. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-09 to explain why they did not include single-failure proof design criteria and ITAAC in Tier 1 of the DCD for the polar crane, the cask handling crane, the containment equipment hatch hoist and the containment maintenance hatch hoist.

In its response dated September 3, 2008, the applicant stated that the single-failure proof criteria within ITAAC Table 2.3.5-2 for components 3a (polar crane), 3b (cask handling crane), 3c (equipment hatch hoist), and 3d (maintenance hatch hoist) will be incorporated.

The applicant also provided proposed revisions to DCD Revision 16, Tier 1, Section 2.3.5, Paragraph 3 under the heading “Design Description,” to state the containment polar crane, cask handling crane, equipment hatch hoist and maintenance hatch hoist are single-failure proof. Also, in Tier 1 Table 2.3.5-2, the applicant proposed changes to ITAAC for the polar crane, the cask handling crane, the equipment hatch hoist and the maintenance hatch hoist.

Based on its evaluation, the staff finds that the applicant demonstrated in the proposed revisions that single-failure proof design criteria and ITAAC are identified in Tier 1 of the DCD for the polar crane, cask handling crane, containment equipment hatch hoist and the containment maintenance hatch hoist. The staff finds the applicant’s response acceptable in that the applicant has now included single-failure proof design criteria and ITAAC in Tier 1 of the DCD for the four load handling systems. The staff confirmed that the applicant made the changes in DCD Tier 1 Revision 17, as described above.

However, the staff finds that the acceptance criteria for the proposed ITAAC should include, not only a report that concludes the acceptability of the proposed inspections, tests, and analyses, but also a certificate of conformance from the vendor stating that the crane/hoist is single-failure proof.

To address the staff's concern regarding the certification of the cranes and hoists as single-failure proof, the applicant provided the staff with Revision 1 to its response to RAI-SRP9.1.5-SBPB-09 in a letter dated June 4, 2009. In the revised response, the applicant stated that single-failure proof criteria will be updated in DCD Tier 1 ITAAC Table 2.3.5-2 for Design Commitments 3a, 3b, 3c and 3d, which are for the two cranes and two hoists. The proposed acceptance criteria for the ITAAC to be incorporated into the next DCD revision will include a certificate of conformance from the crane or hoist vendor stating that the crane or hoist is single-failure proof. The staff finds the response acceptable since the requirement for the crane or hoist vendor to provide a certificate of conformance for being single-failure proof will be added to the acceptance criteria for the crane or hoist ITAAC.

The staff's concern described in RAI-SRP9.1.5-SBPB-09 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.5.2.2 Codes and Standards for Design of OHLHS Cranes and Hoists

This design change corrects wording to state that the polar crane and cask handling crane are designed in accordance with the guidelines of NUREG-0554 supplemented by ASME NOG-1 for a Type I, single-failure proof crane. This change made the DCD consistent with NUREG-0800 Section 9.1.5. Therefore, the staff finds this design change acceptable.

However, for the single-failure proof equipment hatch hoist and maintenance hatch hoist, the applicant only specified ASME NOG-1 and the applicable ANSI standard. ASME NOG-1 can specify non single-failure proof components in the Type II and Type III designs. DCD Revision 16 also stated under Section 9.1.5.2, that the containment equipment hatch hoist and maintenance hatch hoist incorporate single-failure proof features based on NUREG-0612 guidelines. Unlike the polar crane and cask handling crane, there were no detailed descriptions and no specific sections of ASME NOG-1 specified for the containment equipment hatch hoist and maintenance hatch hoist in DCD Section 9.1.5 to make them single-failure proof. Therefore, the staff found the classification for the single-failure proof hoists did not state all necessary design specifications for a single-failure proof component and requested the applicant in RAI-SRP9.1.5-SBPB-01 to provide more design specifications for single-failure proof hoists and a description of the hoists in the DCD.

In its response dated June 26, 2008, the applicant stated that the design specification for the maintenance hatch hoist system and equipment hatch hoist system would follow the guidelines of NUREG-0554, supplemented by ASME NOG-1. The applicant further stated that Table 3.2-3 of the DCD would be revised to reflect these guidelines.

In DCD Revision 17, Tier 2, Table 3.2-3 (Sheet 8 of 65), the applicant revised the table to show the principal construction code for the maintenance hatch hoist and equipment hatch hoist as NUREG-0554, supplemented by ASME NOG-1. However, although the applicant revised DCD Table 3.2-3 to reflect the principal construction code for the two hoists as NUREG-0554, supplemented by ASME NOG-1, the staff noted that the text in DCD Sections 9.1.5.1.2 and 9.1.5.2 has not been changed to reflect the change in the two hoists' construction code, which are NUREG-0554, supplemented by ASME NOG-1 <u>for a Type 1 single-failure proof hoist</u>.

To address the staff's concern, the applicant provided Revision 1 to its response to RAI-SRP9.1.5-SBPB-01 in a letter dated April 13, 2009. The applicant stated that ASME NOG-1, Type 1 designation is not applicable for equipment hatch hoists and maintenance hatch hoists, as it applies to the design of overhead and gantry cranes from the rails to the load hook. The applicant further proposed that the single-failure proof hatch hoists are designed, fabricated, examined and tested in accordance with cmAA 70 and the guidelines of NUREG-0554, supplemented by provision of ASME NOG-1 as it relates to single-failure proof hoists. The staff found the applicant's Revision 1 response to RAI-SRP9.1.5-SBPB-01 unacceptable. In the response, the applicant stated that the requirements of ASME NOG-1, Type 1 do not apply to equipment hatch hosts and maintenance hatch hoists, but the applicant proposed to apply portions of the NOG-1 requirements anyway. Since the applicant proposed to apply only portions of the applicable single-failure proof criteria to the hatch hoists, the staff asked the applicant to clearly define which portions of NRC guidance (e.g., NUREG-0554 and NUREG-0612) and ASME NOG-1 are applied to the heavy load handling cranes in order to classify them as single-failure proof cranes. The staff also asked the applicant to clarify the ambiguous use of the term "NUREG-0554 supplemented by ASME NOG-1," for all of the single-failure proof cranes. It is unclear whether all of ASME NOG-1, Type I, is applied to the polar and cask handling cranes. Revision 15 of the DCD clearly defined the polar crane as designed to ASME NOG-1 for a Type I, single-failure proof crane. However, DCD Revision 17 indicates "NUREG-0554 supplemented by ASME NOG-1." In RAI-SRP9.1.5-SBPB-01, the applicant was asked to provide justification and clearly define how the proposed polar crane, cask handling crane and hatch hoists designs would satisfy the single-failure proof criteria from the NRC guidance and industry standards with references to the specific paragraphs. In RAI-SRP9.1.5-SBPB-01, the applicant was also requested to provide an evaluation of the selected hoist standard to applicable portions of NUREG-0554 to avoid consideration of potential load drops by classification of the hoist as single-failure proof. This issue was tracked as Open Item OI-SRP9.1.5-SBPB-01 in the SER with open items. To address the staff's concern, the applicant provided a Revision 2 to its response to RAI-SRP9.1.5-SBPB-01 in a letter dated November 11, 2009. The applicant's response provided a detailed description of the hatch hoists to include in DCD Sections 9.1.5.2.3 and 9.1.5.2.4. These sections provide specific design criteria of the foot-mounted equipment hatch hoist, with content similar to what is specified for the polar crane and cask handling cranes. A revision to Section 9.1.5.2.3.2, "Component Descriptions," was proposed to incorporate the hoists description and elaborate on how the code requirements are implemented in the design of key safety-related components. The staff found finds the incorporation of the hatch hoist information in the DCD to be acceptable.

However, based on the applicant's clarification that the hatch hoists are foot-mounted on a platform supported by the containment structure, the applicant needed to justify how the structural loads on containment from the hoist are evaluated. For the proposed single-failure proof hatch hoists, there was no assurance that the evaluation considered the lifted load in conjunction with the seismic accelerations because the previous non-single-failure proof hoist was not required to hold a load during and following a seismic event. This became Open Item OI-SRP9.1.5-SBPB-01 in the SER with open items.

The applicant provided a response to Open Item OI-SRP9.1.5-SBPB-01 on March 31, 2010 describing the methodology for its seismic evaluation of the equipment hatch hoist. The response defined the acceptance criteria as "after a seismic event occurs while the hoist is holding the critical load, the containment vessel will continue to perform its intended safety functions." On May 19, 2010, the applicant provided an additional response to Open Item OI-SRP9.1.5-SBPB-01 indicating that the final loads were expected to be bounded by the current CV Design Specification. As a follow-up, the applicant provided an additional Open Item OI-SRP9.1.5-SBPB-01 response dated June 30, 2010, which indicated that the calculation number APP-MH40-S2C-002 includes the hatch hoist platform structural analysis and that the applicant has included the hatch hoist configuration in its structural analysis.

On August 16, 2010, the staff conducted a regulatory audit review of the AP1000, “Hatch Hoist Platform Structural Qualification,” (APP-MH40-S2C-002). In this design analysis, the applicant constructed a 3D finite element model for the platform structural analysis using a commercially available general purpose code (ANSYS). The platform model consists of a 3D beam (including rigid beams), plate/shell elements and lumped masses to represent both the platform and the hatch hoist system. The methodology, input/output and boundary conditions used were reviewed and found acceptable. Three loading cases were considered: (1) dead weight (1 g); (2) dead weight plus seismic; and (3) dead weight minus seismic. The maximum stresses for each structural component are obtained from the equivalent static analysis and used for design purposes. It was shown that use of 8x8x 5/16 beam and 10x10x5/16 beam with an E70xx fillet weld of 0.635 cm (0.25 in) thickness at the connection to the reactor vessel wall is in compliance with ANSI/American Institute of Steel Construction (AISC) N690-1994, “Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities (SC-I),” the calculation is acceptable as it provides assurance of an adequate design to support the heavy hatch hoist system under design-basis seismic loading conditions.

Based on the above review, the staff considers Open Item OI-SRP9.1.5-SBPB-01 resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.5.2.3 Remote Control Operation Change for Polar Crane and Cask Handling Crane

In TR-106 Revision 1, the applicant states that allowing the cask handling crane to be operated by radio remote control instead of the operator’s cab will allow for an unobstructed view of the load at all times. Special consideration was given to loads being lifted and lowered out of and into the truck/rail bay. The cask handling crane radio remote will meet ASME NOG-1 paragraph 6110 guidelines. TR-106 Revision 1 also states that wording for the polar crane operation is changed from "pendant controls" to "remote control" as a secondary means of control. This would ensure consistency in design and operations of the two single-failure proof cranes (polar and cask handling). The staff reviewed TR-106 Revision 1 specifically for the addition of the radio remote control for the cask handling crane and remote control for the polar crane, and found the additions acceptable. The design change is in compliance with GDCs 4, 5, 13, and 24.

However, the licensing basis for the remote control features of the polar crane and the cask handling crane needs to be established in the DCD along with an appropriate ITAAC to verify that the plant meets the licensing basis. The DCD should include information indicating that: a) the remote control features of the polar crane and cask handling crane will not interfere with any SSC important to safety in accordance with GDC 4; b) the remote control systems will not be shared with multiple unit sites or interfere with other units at the site in accordance with GDC 5; c) remote control systems will maintain variables and systems within prescribed operating ranges in accordance with GDC 13; and d) remote control systems will be separate from protection systems, such that failure of the control system leaves the protection systems intact satisfying all reliability, redundancy, and independence requirements in accordance with GDC 24.

In RAI-SRP9.1.5-SBPB-12, the staff requested that the applicant specify the licensing basis for the remote control features of the polar crane and the cask handling crane in the DCD and establish ITAAC to verify the plant meets the licensing basis.

In its response to RAI-SRP9.1.5-SBPB-12, dated January 12, 2009, the applicant addressed each of the licensing basis questions by stating: a) the remote control system will comply with Section 8, Electromagnetic Compatibility (EMC Qualifications) of the Equipment Methodology (EQ) plan, APP-GW-G1-002, Revision 1. The transmitter power of the remote control system for the two cranes will be set to a level that will allow continuous communications with the crane receivers throughout their areas of operation. The signal strength will be adjusted to minimize signal propagation to areas outside of the two cranes normal operating areas; b and c) the two cranes' remote control systems will be designed so that they are integrated into the operating site's frequency plan; and d) the remote control systems will be designed to fail into a safe mode of operation in the event of a loss of communications or failure of the portable remote unit. Should a system failure occur, all crane movements are halted.

The staff finds the applicant's response acceptable since the addition of the remote control systems to the polar and cask handling cranes does not impact the conclusions made in this report about the heavy load handling systems. The design and operation of the actual crane portions of the polar and cask handling cranes does not change with the use of radio remote controls as the controls are the same as the pendant (cask handling crane) and cab controls (polar crane). Therefore, the applicant does not need to provide additional licensing basis and ITAAC verification.

The applicant updated DCD Sections 9.1.5.2.1 and 9.1.5.2.2.1 in Revision 16 to include the use of radio remote control for the polar and cask handling cranes. No further DCD updates are necessary for remote control design changes/additions, and the staff's concern described in RAI-SRP9.1.5-SBPB-12 is resolved.

9.1.5.2.4 Upgrade of Cask Handling Crane to Single-failure Proof and Detailed Description Addition

TR-106 Revision 1 states the AP1000 cask handling crane design was neither seismically qualified nor single-failure proof to protect against dropping a cask. Since a cask drop could cause significant plant damage, a design change was initiated to upgrade the cask handling crane to single-failure proof. This change was made to reduce the possibility of a serious plant event.

With this design change, the applicant revised the name of the crane from the spent fuel shipping cask crane to the cask handling crane and upgraded it to single-failure proof. TR-106 Revision 1 provided a detailed description of the new single-failure proof cask handling crane which was added to DCD Revision 17. Because both the polar crane and cask handling crane are single-failure proof, the added detailed description of the cask handling crane is very similar to the polar crane detailed description. Because the design and system operation of the cask handling crane is described almost identically to that of the polar crane, which was found to be in compliance with GDC 2, 4, 5 and 61, the staff finds this design change of adding a detailed description of the cask handling crane to the AP1000 OHLHS acceptable.

NUREG-0800 Section 9.1.5 and NUREG-0612 provide guidance that states that safe load paths should be defined for movement of heavy loads. However, the applicant had not provided procedures and equipment layout drawings that show safe load paths for movement of heavy loads to minimize the potential to impact irradiated fuel in the reactor vessel and in the SFP and safe shutdown equipment. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-05 to provide equipment layout drawings that show safe load paths and to provide a COL information item for COL applicants to provide procedures that define safe load paths.

In its response dated June 26, 2008, the applicant stated that the equipment layout drawings that show safe load paths are not provided in the DCD. This information is part of the operational programs and is covered by Section 13.4 of the DCD. The staff finds that Section 13.4 does not necessarily state that the COL applicant will develop procedures that show safe load paths. In evaluating the response, the staff determined that the applicant had not complied with the guidelines specified in NUREG-0800 Section 9.1.5 and NUREG-0612 as stated in that safe load paths had not been identified. Since the AP1000 is a standard design nuclear power plant where the location of all stationary equipment in non-site specific structures has been determined, generic safe load path figures should be developed for known heavy load lifts, such as the reactor vessel head and the spent fuel cask by the applicant for the AP1000 design.

To address the staff's concern, the applicant provided Revision 1 to its response to RAI-SRP9.1.5-SBPB-05, dated April 13, 2009. The applicant stated that a COL information item would be incorporated into the DCD requiring COL applicants to provide a heavy load handling program. The applicant proposed to modify DCD Section 9.1.5 to include a statement pointing out that DCD Section 13.5.1 addresses the development of heavy lift safe load paths. Following the guidance in NUREG-0800 Section 9.1.5 and NUREG-0612, the applicant proposed to modify DCD Section 13.5.1 to include the information that COL applicants referencing the AP1000 certified design would provide a heavy load handling program, which would include safe load paths for movement of heavy loads, to be referenced in procedures and shown on equipment layout drawings. DCD Section 13.5.1 includes the statement that the program and associated procedures will minimize the potential to impact irradiated fuel in the reactor vessel and in the SFP, and safe shutdown equipment from movement of heavy loads. The applicant also stated that it is currently developing drawings identifying safe load paths for the handling of heavy loads, which will then be provided to COL applicants for incorporation into their heavy load handling programs.

The applicant indicated that the revisions to DCD Tier 2 Sections 9.1.5 and 13.5.1 requiring that a COL applicant referencing the AP1000 certified design would provide a heavy load handling program, including safe load paths, would be added to the next DCD revision.

The staff finds the response acceptable since the applicant is developing standard safe load path drawings for heavy loads, which would be provided to the COL applicants. The applicant also added a COL information item for COL applicants to provide a heavy load handling program, which includes safe load paths to be referenced in procedures and shown on equipment layout drawings. It should be noted that the applicant has not created a completely new COL information item but added the requirement for a COL applicant to provide a heavy loads program to existing COL Information Item 13.5.1, which requires COL applicants to provide plant procedures.

The staff's concern described in RAI-SRP9.1.5-SBPB-05 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

NUREG-0800 Section 9.1.5 and NUREG-0612 provide guidance for applicants to describe a heavy load handling program for design, operation, testing, maintenance and inspection of heavy load handling systems. The applicant had not provided a heavy load handling program. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-06 to provide a COL information item to ensure that the COL applicant will provide such a heavy load handling program.

In its response dated June 26, 2008, the applicant stated that it would provide the COL holders with the Operations and Maintenance Manuals for heavy load handling systems so that the manuals could be used when they created their programs. The applicant further stated that operations programs and procedures were discussed in Sections 13.4 and 13.5 of the DCD to include existing COL information items and no further COL Information Items were necessary. The staff determined that Sections 13.4 and 13.5 did not specify that the COL applicant would provide the heavy load handling program elements specified in NUREG-0800 Section 9.1.5 Section III.3 and NUREG-0612 Section 5.1.1. Therefore, the staff asked the applicant to revise the DCD to include a COL information item similar to RG 1.206, “Combined License Applications for Nuclear Power Plants,” Regulatory Position Part III, Section C.I.9.1.5.

To address the staff’s concern, the applicant provided Revision 1 to its response to RAI-SRP9.1.5-SBPB-06, dated April 13, 2009. The applicant’s revised response states that this concern is addressed in the applicant’s RAI-SRP9.1.5-SBPB-05, Revision 1 response.

The staff finds the applicant’s RAI-SRP9.1.5-SBPB-05, Revision 1 response, as discussed above, adequately addresses the addition of a DCD COL information item requiring a COL applicant to develop a heavy loads handling program. Therefore, RAI-SRP9.1.5-SBPB-06 is resolved.

In DCD Revision 17, Section 9.1.5.2.2.2, “Component Descriptions,” under the heading “Lifting Devices Not Specially Designed,” (for the cask handling crane), the applicant states that for the handling of <u>critical</u> loads, dual or redundant slings are used, or a sling having a load rating twice that required for a <u>non-critical</u> load is used. Therefore, the staff requested in RAI-SRP9.1.5-SBPB-02 that the applicant explain how the statements are in conformance with the NUREG-0612 criteria, when a sling having a load rating twice that required for a <u>non-critical</u> load is used instead of a load rating twice that required for a <u>critical</u> load. This same issue is in Section 9.1.5.2.1.2 of DCD Revision 16.

In its response dated June 26, 2008, the applicant stated that in selecting the proper sling for rigging, the load capacity of the sling should be greater than the sum of the maximum static and dynamic load to be lifted. The applicant then stated that no matter what type of load is being lifted “critical lift or non-critical lift,” the sling rating needs to take into account both static and dynamic loading. The response stated:

> But when you are making a “critical lift” then you must either:
>
> 1) Use two (2) of the properly rated slings per the formula above [the summation of the static and dynamic loads], OR
>
> 2) Use one (1) sling with twice (2x) the proper rating per the formula above.

In evaluating the response, the staff finds the applicant has clearly stated the application of the requirements of NUREG-0612 for selecting slings for lifting critical loads. The applicant has properly stated that for critical loads, redundant slings must be used or the selected sling must have a load rating of twice the minimum required capacity. In DCD Revision 17, the applicant elected to use the term “non-critical load” in making a comparison between the normal selection of a sling for lifting a load and the selection of a sling or redundant slings under NUREG-0612 for lifting a load that has been designated a critical load. A “non-critical” load would require only a single sling rated for the summation of the static and dynamic load required to be lifted. However, if the same load was declared a “critical” load, the capacity of the single sling would have to be doubled, or twice the capacity required for the “non-critical” load. Based on its review, the staff finds the applicant’s clarification of the DCD text acceptable and RAI-SRP9.1.5-SBPB-02 is resolved.

In DCD Revision 16, Sections 9.1.5.2.1.2 and 9.1.5.2.2.2, both titled “Component Descriptions,” the applicant stated under the headings “Lifting Devices Not Specially Designed” that slings or other lifting devices not specially designed are selected in accordance with ANSI B30.9, “Slings,” except that the load rating is based on the combined maximum static and dynamic loads that could be imparted to the sling. The two separate headings are associated with the polar crane and cask handling crane, respectively. NUREG-0800 Section 9.1.5 Revision 1, “Overhead Heavy Load Handling Systems,” states in Paragraph III.4.C.ii (2), that slings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope). Therefore, the staff requested the applicant, in RAI-SRP9.1.5-SBPB-03, to explain how the criterion that slings be constructed of metallic material (chain or wire rope) for single-failure proof cranes and hoists that are handling critical loads is satisfied.

In its response dated June 26, 2008, the applicant stated that in addition to the polar crane and cask handling crane, the maintenance hatch hoist and equipment hatch hoist use lifting devices not specially designed that meet the safety factor requirements of ASME B30.26 2004, “Rigging Hardware.” The applicant also stated the design specifications for the polar crane and cask handling crane reference ASME B30.9, “Slings,” and the NRC Regulatory Issue Summary (RIS) 2005-25, Supplement 1, “Clarification of NRC Guidelines for Control of Heavy Loads,” and that these references were added to the DCD. The supplement states that: Slings should satisfy the criteria of ASME B30.9-2003, “Slings,” and be constructed of metallic material (chain or wire rope). Revision 17 of DCD Sections 9.1.5.2.1.2 and 9.1.5.2.2.2 was revised to indicate that the slings shall be constructed of metallic material (chain or wire rope) per NRC RIS 2005-25, Supplement 1.

The staff finds the applicant committed to using slings made from metallic material when lifting critical loads. The applicant demonstrated that it is in compliance with NUREG-0800 Section 9.1.5, Revision 1, Paragraph III.4.C.ii (2), which specifies that slings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope). Therefore, the staff finds the applicant’s response to be acceptable and the staff’s concern described in RAI-SRP9.1.5-SBPB-03 is resolved.

In DCD Revision 16, Section 9.1.5.2.2.2, “Component Descriptions,” under the heading “Special Lifting Devices” (for the cask handling crane), the applicant stated that the special lifting devices used for the handling of critical loads are listed in Table 9.1.5-2. The staff’s review of Table 9.1.5-2 finds only special lifting devices for the polar crane and none for the cask handling crane. Existing plant operating experience demonstrates that a special lifting device is normally used between a cask and the cask handling crane hook due to the shape and size of the cask. Therefore, the staff requested the applicant in RAI-SRP9.1.5-SBPB-04 to explain if a special lifting device will be used between the cask and cask handling crane hook and if so, why it is not listed in Table 9.1.5-2. If a special lifting device is not used, the staff asked the applicant to explain the anticipated rigging of the cask to the cask handling crane hook.

In its response dated June 26, 2008, the applicant stated that special lifting devices will be used with the cask handling crane and will be added to the DCD in Table 9.1.5-2. The applicant, in its response, provided a markup of DCD Table 9.1.5-2, "Special Lifting Devices Used for the Handling of Critical Loads." The applicant added the following special lifting devices to Table 9.1.5-2 for the cask handling crane: cask lift yoke, cask lift yoke extension, and loaded canister handling equipment. The following description was also added to the table for these devices: These devices are used for the handling of the casks and loaded canisters, which provide the interface between the cask handling crane and the shipping cask or loaded canister.

In DCD Tier 2, Table 9.1.5-2 Revision 17, the applicant revised the table to show the special lifting devices to be used with the cask handling crane and also provided a description of the use of these devices. With the addition of the cask handling crane special lifting devices to DCD Table 9.1.5-2, the staff finds the applicant's response acceptable and the staff's concern described in RAI-SRP9.1.5-SBPB-04 is resolved.

9.1.5.2.5 Increase in Polar Crane Maximum Load Rating

TR-106 Revision 1 states the critical lift for the polar crane is the lifting of the Integrated Head Package from the Reactor Vessel to the in-containment storage stand during a refueling outage. The critical lift weight for the polar crane has been increased from 249.48 metric tons (275 tons) to 272.16 metric tons (300 tons) to ensure adequate lifting margin. Because the weight of the integrated head package is the critical lift for the polar crane that requires the polar crane to have its maximum load rating increased to 272.16 metric tons (300 tons), while still being a single-failure proof crane in compliance with GDC 2, 4, 5 and 61, the staff finds this design change to be acceptable.

9.1.5.2.6 Addition of Cask Handling Crane Component Data

TR-106 Revision 1 states the AP1000 cask handling crane design was neither seismically qualified, nor single-failure proof to protect against dropping a cask. Since a cask drop could cause significant plant damage; a design change was initiated to upgrade the cask handling crane to single-failure proof. This change was made to reduce the possibility of a serious plant event.

With this design change the crane name was changed from the spent fuel shipping cask crane to the cask handling crane. With the changing of the design of the cask handling crane (previously spent fuel shipping cask crane in DCD Revision 15) to single-failure proof in DCD Revision 17, Table 9.1.5-1 has been completely changed to provide cask handling crane component data. The previous information in Table 9.1.5-1 under Revision 15 was for a non-single-failure proof spent fuel cask crane. DCD Revision 17 Table 9.1.5-1 provides design information for the cask handling crane bridge, trolley, main hoist and auxiliary hoist. Because the addition of the component data for the cask handling crane in Table 9.1.5-1 is similar to the data provided in Table 9.1.5-3 for the polar crane and does not affect the single-failure proof cask handling crane's compliance with GDC 2, 4, 5 and 61, the staff finds this design change to be acceptable.

9.1.5.2.7 Polar Crane Main Hook Used to Lift RCP

TR-106 Revision 1 states the main hook on the polar crane, instead of the auxiliary hook of the polar crane, will be used to install and remove the RCPs. Since the main hook has a larger load capacity than the auxiliary hook and is single-failure proof, the applicant remains in compliance with GDC 4. Therefore the staff finds this design change to be acceptable.

9.1.5.2.8 Decrease in Polar Crane Auxiliary Hook Capacity

TR-106 Revision 1 states the auxiliary hook on the polar crane has been reduced from 68.04 metric tons (75 tons) to 22.68 metric tons (25 tons) now that it is no longer being used to install and remove the RCPs. Because the polar crane main hook will now be used with the RCP special lifting device instead of the auxiliary hook, the capacity of the auxiliary hook can be reduced from 68.04 metric tons (75 tons) to 22.68 metric tons (25 tons). The staff finds that with the change in lift capacity of the auxiliary hook, the polar crane remains a single-failure proof crane and in compliance with GDC 2, 4, 5 and 61.

9.1.5.2.9 Additional Staff Inquiries Associated With Overhead Heavy Load Handling

In RAI-SRP9.1.5-SBPB-07, the staff asked the applicant the following six clarifying questions to provide a better understanding of the applicant's submitted technical information:

a) In RAI-SRP9.1.5-SBPB-07, Subpart a, the staff asked the applicant to clarify if the cask handling crane was to be operated from either a radio remote control, operator's cab or a pendant suspended from the crane since it was not totally clear after a review of the information in TR-106 Revision 1.

In its response dated June 26, 2008, the applicant stated that the cask handling crane will be operated by radio remote control or from a pendant suspended from the crane and that the crane does not have an operator's cab.

Based on its review, the staff finds the applicant's response clarifies the wording in TR-106 and is consistent with DCD Revision 17, in that the applicant clarifies that the cask handling crane will be operated by either radio remote control or from a pendant suspended from the crane. The applicant also updated DCD Sections 9.1.5.2.1 and 9.1.5.2.2.1 to include the use of radio remote control for the polar and cask handling cranes. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart a is resolved.

b) In RAI-SRP9.1.5-SBPB-07, Subpart b, the staff asked the applicant to clarify why in TR-106 Revision 1 for Design Change 256 the title of the design change is for the maintenance hatch hoist design while the cask handling crane is mentioned twice in the text explaining the design change.

In its response dated June 26, 2008, the applicant provided an excerpt from TR-106 Revision 1 of the text from Design Change 256 with the cask handling crane words crossed out in two places and replaced with the words Maintenance Hatch Hoist. No technical information that would affect the DCD was changed.

Based on its review, the staff finds the applicant's response acceptable because the applicant proposed corrections to the identification of the crane in TR-106 Revision 1, Design Change 256 from the cask handling crane to the maintenance hatch hoist; making the design change text agree with the design change title in TR-106. Therefore, no change to the DCD was needed and the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart b is resolved.

c) In RAI-SRP9.1.5-SBPB-07, Subpart c, the staff asked the applicant to provide the correct section to which a reader is referred in the paragraph under Section 9.1.5.2, System Description, of AP1000 DCD Revision 16, since the currently referenced section does not exist.

In its response dated June 26, 2008, the applicant stated the correct referenced section is 9.1.5.3, not 9.1.5.2.3, and provided a markup of paragraph 9.1.5.2 from DCD Revision 16 showing the correction.

In DCD Tier 2, DCD Section 9.1.5.2 Revision 17, the applicant revised the referenced subsection to show 9.1.5.3. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart c is resolved.

d) In RAI-SRP9.1.5-SBPB-07, Subpart d, the staff provided an excerpt from TR-106, Revision 1, Design Change Description for Design Change 170 (polar crane Design) which states: The critical lift for the polar crane is the lifting of the integrated head package (IHP) from the reactor vessel head to the in-containment storage stand during a refueling outage. The staff asked the applicant to clarify if the IHP is lifted by the polar crane from the reactor vessel head or from the reactor vessel, which components comprise the IHP, and if the IHP and reactor vessel head are two separate lifts by the polar crane.

In its response dated June 26, 2008, the applicant stated that the IHP is lifted by the polar crane from the reactor vessel. The IHP includes the reactor vessel head, the shield shroud, the control rod drive mechanism cooling fans, and the lifting rig. The IHP and the reactor vessel head are not two separate lifts.

Based on its review, the staff finds the applicant's response acceptable because the applicant clarified that the IHP is not lifted from the reactor vessel head as stated in TR-106, Revision 1, Design Change Description for Design Change 170. The IHP is lifted from the reactor vessel by the polar crane. The IHP and the reactor vessel head are not two separate lifts and the IHP is properly defined in Section 3.9.7 of the DCD. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart d is resolved.

e) In RAI-SRP9.1.5-SBPB-07, Subpart e, the staff asked the applicant why the title for Section 9.1.6.5 of the AP1000 DCD Revision 16 was "Inservice Inspection Light Load Handling System," and did not include the OHLHS.

In its response dated June 26, 2008, the applicant stated the title for Section 9.1.6.5 should be changed to "Inservice Inspection Load Handling System" from "Inservice Inspection Light Load Handling System." Based on its review, the staff finds the applicant's response acceptable because the applicant proposes to revise the title of DCD Section 9.1.6.5 to cover both light and heavy load handling systems.

In DCD Tier 2, DCD Section 9.1.6.5 Revision 17, the applicant changed the name of this section from Inservice Inspection Light Load Handling Systems to Inservice Inspection Load Handling Systems. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart e is resolved.

f) In RAI-SRP9.1.5-SBPB-07, Subpart f, the staff stated that the cask handling crane was still referred to as the spent fuel shipping cask crane in Tier 1 Section 2.3.5, Item Number 4 of AP1000 DCD Revision 16 and the section needed correction.

In its response dated June 26, 2008, the applicant states the cask handling crane should not be referred to as the spent fuel shipping cask crane and the DCD needs to be changed in four sections to make the correction. The applicant provides two excerpts from Tier 1 and two excerpts from Tier 2 of the DCD showing where the words spent fuel shipping cask had been crossed out and replaced with cask handling. Based on its review, the staff finds the applicant's response acceptable because the applicant corrected the name of the cask handling crane from spent fuel shipping cask crane in four sections of DCD Revision 17. In DCD Tier 1, Revision 17, Section 2.3.5 and Table 2.3.5-2, the applicant changed the name of the previously shown spent fuel shipping cask crane to cask handling crane. In DCD Tier 2, Revision 17, Sections 9.1.5.3 and 14.2.9.4.14, the applicant also changed the name of the previously shown spent fuel shipping cask crane to cask handling crane. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-07, Subpart f is resolved.

In RAI-SRP9.1.5-SBPB-10, the staff requested additional information related to heavy load drop analysis. In the AP1000 DCD Tier 2, Revision 16, Section 9.1.5.3, "Safety Evaluation," the applicant states: "Postulated load drops are evaluated in the heavy load analysis." In the last sentence of Section 9.1.5.3 the applicant states, "The heavy load analysis is to confirm that a postulated load drop analysis does not cause unacceptable damage to reactor fuel elements, or loss of safe shutdown or decay heat removal capability." The staff asked the applicant to describe what heavy load drop analyses were performed and to describe the results of the analyses.

In its response dated September 3, 2008, the applicant stated that the polar crane, cask handling crane, equipment hatch hoist, and maintenance hatch hoist are single-failure proof, which satisfies the requirements for moving heavy loads, and no heavy load drop analysis was performed. The applicant added single-failure proof criteria to ITAAC Table 2.3.5-2 in DCD Revision 17 for all four of these load handling systems, as identified in its response to RAI-SRP9.1.5-SBPB-09. Additionally, the applicant stated that the main steam isolation valve (MSIV) monorail hoists A and B are used to perform maintenance on the MSIVs. However, the hoists will not be used during plant operation and, therefore, failure of the hoists (while lifting loads) will not prevent the plant from shutting down safely because the plant will already be shut down.

Based on its review, the staff finds acceptable the part of the applicant's response, that states that no heavy load drop analyses are required for the polar crane, cask handling crane, equipment hatch hoist, and maintenance hatch hoist because they are designed single-failure proof. However, the staff determined that the response was unacceptable regarding the MSIV monorail hoists A and B, because the applicant did not address the effect of a load drop on equipment needed for decay heat removal.

To address the staff's concern, the applicant provided Revision 1 to its RAI-SRP9.1.5-SBPB-10 response, dated January 28, 2009. The applicant stated the plant modes in which the main steam isolation values must be operable. Because the MSIVs and main steam safety valves (MSSVs) have to be operable or closed during plant modes 1, 2, 3, and 4, the MSIV monorail hoists shall not be used to service the MSIVs or MSSVs during Modes 1, 2, 3, or 4. The applicant further stated that during Modes 5 and 6 the steam generators are not utilized for nonsafety-related decay heat removal and the MSIVs could be taken out of service. Therefore, a load drop by the MSIV monorail hoists during modes 5 or 6 would not affect decay heat removal capability of the AP1000.

Based upon its review, the staff determined that the response was unacceptable regarding the MSIV monorail hoists A and B, because the applicant did not address the effect of a load drop on equipment needed for decay heat removal.

To address the staff's concern, the applicant provided Revision 2 to its RAI-SRP9.1.5-SBPB-10 response, dated April 13, 2009. The applicant's revised response states that equipment and components required for decay heat removal during Modes 5 or 6 are not located in the load path for the MSIV monorail hoists.

In the revised response, the applicant proposed revisions to DCD Tier 2 Section 9.1.5.3 that explicitly state that the equipment and components required for decay heat removal during Modes 5 or 6 are not located in the load path of the MSIV monorail hoists.

The staff finds the response acceptable since the applicant clarified that equipment and components required for decay heat removal during Modes 5 or 6 are not located in the load path for the MSIV monorail hoists. The staff's concern described in RAI-SRP9.1.5-SBPB-10 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.1.5.2.10 Staff Inquiry Regarding Design Changes in TR-106 not reflected in DCD Revision 16

In RAI-SRP9.1.5-SBPB-08, the staff stated that TR-106, Revision 1 in Section V described post AP1000 Revision 16 changes. The staff provided paragraphs a through d of TR-106, Section V showing the post DCD Revision 16 changes and asked the applicant to verify if the changes would be documented in the DCD.

In its response dated June 26, 2008, the applicant stated that all the post AP1000 Revision 16 design changes described in TR-106, Revision 1, Section V would be incorporated in Revision 17 of the DCD.

In Revision 17 of the DCD Tier 2, Sections 9.1.5.2.1.3, 9.1.5.2.2.3 (both sections titled "Instrumentation Applications"); 9.1.5.3, "Safety Evaluation": Table 9.1.5-1, "Cask Handling Crane Component Data"; and Table 9.1.5-3, "Polar Crane Component Data," the applicant has incorporated the changes shown in TR-106 Revision 1 Section V.

In the three DCD Tier 2 sections and two DCD Tier 2 Tables listed above, the applicant proposed changes to align DCD Section 9.1.5 and the associated tables with ASME NOG-1 and NUREG-0554 for single-failure proof cranes.

Specific DCD Revision 17 changes include:

1. In DCD Section 9.1.5.3 the applicant stated that either redundancy or double design factor for load bearing components such as the hoisting ropes, sheaves, equalizer assembly, hooks and holding brakes of single-failure proof cranes is permitted.

The staff determined that the proposed change for potentially allowing redundancy for all load bearing components was unacceptable because it describes a non-redundant equalizer device as acceptable and because the new statement regarding which components must be redundant was ambiguous. Every component of the reeving system must be redundant except the rope drum, the upper block, the load block, and the hook (which is part of the load block). Among the components listed in TR-106, only design of the hook to twice the normal design load is acceptable in place of redundant hooks. The staff determined that the reference to NUREG-0554, Paragraph 4.3 in TR-106 does not provide the criteria for alternatives to redundancy in load attachment points; those criteria are in Appendix C to NUREG-0612 and ASME NOG-1.

In a letter dated September 8, 2009, the applicant provided a revised response to RAI-SRP9.1.5-SBPB-08. The applicant provided additional changes to DCD Section 9.1.5.3 to provide additional clarification to comply with ASME NOG-1. The staff finds the proposed changes comply with NOG-1 and are acceptable. Therefore, the staff's concern described in RAI-SRP9.1.5-SBPB-08 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

2. In DCD Tables 9.1.5-1 and 9.1.5-3, the applicant changed wording for the Bridge and Trolley sections to reflect that service, emergency, and parking functions may be performed by a single friction brake.

3. In DCD Sections 9.1.5.2.1.3 and 9.1.5.2.2.3, the applicant stated that the hoist in the raising direction for the polar crane and cask handling crane has block-actuated limit switches, which directly interrupt power to the hoist motor and cause the hoist brakes to set.

4. In DCD Tables 9.1.5-1 and 9.1.5-3, the applicant added information on control, holding and emergency brakes for the main hoist and auxiliary hoist.

TR-106 Revision 1 states that changes 2, 3 and 4 were made to better align the DCD design data for the single-failure proof polar crane and cask handling crane. The staff finds changes 2, 3, and 4 acceptable, since the guidance in NUREG-0554, as supplemented by ASME NOG-1 for single-failure proof cranes, is followed. In accordance with NUREG-0800 Section 9.1.5, cranes designed to the criteria of NOG-1 for a Type 1 crane are acceptable under the guidelines of NUREG-0554 for construction of a single-failure proof crane.

9.1.5.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for the DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 OHLHS as documented in AP1000 DCD, Revision 17, and in TR-106, Revision 1, which include a design description for the cask handling crane, an upgrade designation of the cask handling crane to seismic Category I and single-failure proof, an upgrade of the maintenance hatch hoist to single-failure proof, a designation of ASME NOG-1 for a Type 1 crane for single-failure proof cranes and hoists, the replacement of pendant control to remote control for the polar crane, and the change in design capacity of the main and auxiliary hooks of the polar crane. The staff concludes that the AP1000 OHLHS design continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criteria of; 10 CFR 52.63(a)(1)(vi), on the basis that they substantially increase overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security; and 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD. Therefore, the staff finds that the proposed changes to the AP1000 OHLHS are acceptable.

9.2 Water Systems

9.2.1 Service Water System

9.2.1.1 Summary of Technical Information

Revision 17 of the AP1000 DCD includes proposed changes to the service water system (SWS) in order to accommodate increased heat loads, establish increased wet and dry bulb temperature limits for the site, and to provide additional design flexibility for COL applicants. The details of these proposed changes are discussed in TR-111, “Component Cooling System and Service Water System Changes Required for Increased Heat Loads,” APP-GW-GLN-111, Revision 0, dated May 2007; TR-108, and TR-103.

In AP1000 DCD Revision 17, the applicant proposed Tier 2 changes associated with SWS Section 9.2.1.2.2 “Component Description – Piping Requirements.” The applicant corrected the referenced code to use for nonmetallic piping from “ASME B31.1” to “ANSI B31.1, Appendix III.”

The applicant proposed a change to Tier 1 Table 5.0-1, “Site Parameters,” to increase the maximum safety coincident wet bulb temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F) and non-coincident wet bulb temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F). The details of this change are discussed in APP-GW-GLE-036, Revision 0. Because this change could impact the SWS performance, it is evaluated below. In Revision 17, the applicant made no changes to DCD Tier 1, Section 2.3.8, “Service Water System.”

9.2.1.2 Evaluation

The regulatory basis for evaluating the SWS is documented in Section 9.2.1, “Station Service Water System,” of NUREG-1793. While the SWS (including heat sink) is nonsafety-related, it is considered to be important to safety because it supports the normal defense-in-depth (DID) capability of removing reactor and spent fuel decay heat, it is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the SWS makes it subject to RTNSS in accordance with the Commission’s RTNSS policy in SECY-94-084 for passive reactor plant designs. The staff’s evaluation of the changes that are proposed focuses primarily on confirming that the changes will not adversely affect safety-related SSCs, the capability of the SWS to perform its DID and RTNSS functions, and the adequacy of ITAAC, test program specifications, and RTNSS availability controls that have been established for the SWS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.1, “Station Service Water System,” as it pertains to these considerations. Acceptability is judged based upon conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.1 (as applicable), and SECY-94-084.

The specific criteria that apply to the changes referred to above include 10 CFR 52.63(a)(1)(iii), which concerns the proposed changes reducing unnecessary regulatory burden while maintaining protection to public health and safety and the common defense and security; and 10 CFR 52.63(a)(1)(vii), which concerns the proposed changes contribution to increased standardization of the certification information.

9.2.1.2.1 SWS Design Changes Required for Increased Heat Loads

As discussed in TR-111, increased CCS heat loads and flow rates have resulted in the need for corresponding increases in SWS flow and pump capacity, pipe and component size, and cooling tower heat dissipation capability and makeup rate. The proposed SWS changes are reflected in the AP1000 DCD, Tier 1, Table 2.3.8-2, ITAAC Design Commitment Item 2 relative to the capability to support plant shutdown and spent fuel cooling; Tier 2, Table 9.2.1-1, “Nominal Service Water Flows and Heat Loads at Different Operating Modes;” and Tier 2, Table 16.3-2, “Investment Protection Short-Term Availability Controls (IPSAC),” Section 2.4, “Service Water System (SWS) – RCS Open,” SR 2.4.1 with respect to the minimum required SWS flow rate. Section 9.2.2 provides an evaluation of the increased CCS heat loads.

<u>Proposed Increases in SWS Flow Rate and Heat Dissipation Capability</u>

The applicant proposed to increase the minimum required SWS flow rate and heat dissipation capability in order to accommodate the higher heat loads that are proposed for the AP1000 plant design. The values listed in Tier 2 of the DCD, Table 9.2.1-1, and the ITAAC specified by Tier 1 of the DCD, Table 2.3.8-2, reflect these changes. The ITAAC requires COL applicants to demonstrate that the SWS design is capable of supporting plant shutdown and spent fuel cooling. The applicant proposed to change the flow rate value in the ITAAC that demonstrates, by testing, that each SWS pump can deliver at least 37,854 Lpm (10,000 gpm) flow to each CCS HX. The applicant also proposed to increase the required heat transfer rate that is specified for each cooling tower cell to be greater than or equal to 1.8×10^8 kilojoules per hour (kJ/h) (1.7×10^8 Btu/h) at a cold water temperature of 32.2 °C (90 °F). The proposed changes in the ITAAC acceptance criteria relative to SWS flow rate and heat load are consistent with the revised values that are reflected in Tier 2, Table 9.2.1-1, and they appear to be acceptable from this perspective. However, the applicant did not specifically identify how the revised values were determined and on what basis they are considered to be appropriate. For example, the applicant did not explain how the maximum heat load was determined and how much margin is afforded by the ITAAC acceptance criterion. The applicant did not compare the available margin with the amount of margin that is needed based on industry experience to accommodate degradation that is anticipated to occur over time and provide necessary flexibility. Also, the applicant did not explain why the bases for these values and the industry experience that is credited applies to all COL applicants.

The applicant was asked, in RAI-SRP9.2.1-SBPA-01, to provide a more detailed description of the basis for the proposed changes relative to SWS flow rate and cooling tower performance in Tier 2 of the DCD. The applicant responded to the staff’s request in a letter dated June 26, 2008. The applicant referred to information in the application that was considered by the staff when preparing this question, but a more detailed description was not provided. The applicant also provided information to explain how cooling tower performance and SWS flow rates would be maintained over time. While this information is useful, no provisions were established to assure implementation by COL applicants. The RAI response did not adequately address RAI-SRP9.2.1-SBPA-01 because more information was needed regarding RTNSS and systems supporting the ability to achieve cold shutdown operations.

During the June 25, 2009 audit, the staff explained that in order to resolve the RAI, the applicant should provide more detail explaining how the design margins were established and the maintenance and testing activities that ensure that the margins are adequately preserved over the life of the plant.

In a letter dated August 31, 2009, the applicant provided additional information specific to the design margins, appended to its original response to RAI-SRP9.2.1-SBPA-01. Additionally, the applicant provided a proposed mark up of DCD Tier 2, Section 9.2.1.2.3.4 that supports the information provided in the additional information.

The response identified that the SWS is designed with sufficient margin to ensure that system flow rates and cooling tower performance will be maintained such that RTNSS and DID functions can be performed over the life of the plant. The response described the following functions that meet criteria for DID functions of the CCS as they relate to the SWS:

- For normal residual heat removal system (RNS) cooling, the SWS and CCS are needed to cool the RNS HXs and pumps during RCS cooldown and cold shutdown operation, in order to avoid actuation of the passive residual heat removal (PRHR) HX. The SWS and CCS also provide cooling to the RNS during refueling operation, to avoid heat up of the water in the refueling cavity.

- For SFS cooling, the SWS and CCS provide cooling to the SFS HXs during all modes of plant operation to prevent heat up and boiloff of water in the SFP.

- For CVS miniflow HX cooling, the SWS and CCS provide cooling to the miniflow HXs of the CVS injection pumps. This allows proper operation of the CVS injection pumps, in order to avoid core makeup tank (CMT) actuation.

- For reduced inventory cooling, the SWS and CCS provide cooling to the RNS HXs and pumps during reduced reactor coolant inventory operation.

The response specified that the SWS also supports a function of the CCS that is important for equipment protection by providing necessary cooling to the RCP external HX to avoid reaching high bearing water temperatures and its resulting RCP trip.

The response also identified the SWS heat loads as requested by the staff. The heat loads include the RCP external HX, the SFS HX, and the CVS miniflow HX. The response stated that the cooling functions for the CCS and SWS are dependent on the temperature supplied by the CCS HX. The response also explained that, of these three functions, the RCP external HX cooling function has the lowest design temperature requirements and, therefore, provides the high CCS design temperature.

The ability of the CCS to meet the RCP cooling requirement was evaluated by considering the design operating mode for the CCS HX, which is normal power operation. The CCS HX overall coefficient of heat transfer (U) and required heat transfer area (A) was computed by ensuring that the high CCS design temperature limit was not exceeded during this design case. The total normal power operating heat duty included the maximum SFS heat duty (which is immediately after refueling), maximum heat dissipation by the RCPs, as well as a 20 percent margin above the maximum central chilled water system (VWS) chiller heat load. The SWS temperature supplied to the CCS HX also included additional margin, since the maximum cooling tower approach temperature was added to the maximum safety wet bulb.

The response explained that the SWS cooling towers are sized for both plant cooldown (higher heat loads) and normal power operation (lower heat loads). The CCS HX design is able to supply cooling water that meets all of the DID and investment protection cooling requirements under the most limiting conditions. The CCS HX specification also includes an additional 10 percent of heat transfer area above the design value to account for fouling and degradation over the HX's operating life. Additional frame length is also included so that 20 percent more than the nominal plate number required to provide the design heat transfer capability can be added to the HX if additional performance is needed.

SWS cooling tower performance is maintained by providing substantial margin in the sizing of the cooling tower. The SWS cooling towers were sized using a peak cooldown heat duty that includes significant conservatism. The cooling towers must be able to remove sufficient heat to cool down the RCS via the RNS, starting 4 hours after reactor shutdown, to cold shutdown conditions 96 hours after shutdown, assuming the persistence of the ambient wet bulb temperature at the maximum normal value of 26.7 °C (80.1 °F). This assumption itself provides substantial margin since the definition of the AP1000 maximum normal temperature value is based on the 1 percent seasonal exceedance wet bulb temperature, which can be experienced for fewer than 30 hours per year. The sizing case assumes that all four RCPs and variable frequency drives (VFDs) are operating at maximum allowable speed, though procedurally only two RCPs and RCP VFDs should run at 50 percent speed (or less) during this condition. Significant margins in the SFS and VWS chiller heat loads were included, as they were for the normal power operating design case. The margin included in these major heat loads, as well as several others, results in approximately 30 percent margin in the SWS cooling tower design heat duty, with respect to the expected heat duty during a realistic plant cooldown. Since the SWS tower cells are sized to meet cooldown time requirements under this extremely conservative operating case, any long term degradation in cooling ability under more realistic heat duties would not prevent the tower from meeting its heat transfer performance requirements.

It should also be noted that the DID and RTNSS functions of RNS cooling during RCS cooldown and RNS cooling during reduced coolant inventory operation require that the CCS provide cooling during these operating modes, and does not require specific temperature limitations or impose defined time to temperature requirements on the CCS and SWS. The CCS and SWS need to be designed to prevent heat up of the RCS if one train of CCS and SWS is not operable. Heat-up needs to be prevented even under maximum normal ambient wet bulb temperature conditions, though the time to cool down can be extended. This capability of the CCS and SWS also ensures that the DID function of the CCS and SWS can be fulfilled even assuming significant degradation in the CCS HX and SWS cooling tower performance, when both trains are operable. Similarly, ambient conditions above the maximum normal wet bulb temperature would not prohibit the CCS and SWS from performing this function, though cooldown times would be extended. The ability of the CCS and SWS to cool the RNS HXs during reduced coolant inventory operation (Modes 5 and 6) is further assured since the heat duty in this mode of operation is significantly reduced. The overall heat duty of the CCS and SWS in this mode is approximately 33 percent of the heat duty at the beginning of normal RNS cooldown,

The SWS pumps are required to provide a nominal flow rate of 39,750 Lpm (10,500 gpm) to cool the CCS HX for all normal operating modes, though a degraded minimum flow rate of

(10,000 gpm) can support decay heat removal from the RNS and SFS systems. In the RAI response, the applicant stated that the SWS flow analysis indicates that the selected SWS pump delivers 39,750 Lpm (10,500 gpm) to the CCS HX for all normal operating modes with all flow resistances (k-values) in the system increased by 10 percent. An additional 7 percent margin in pump-developed head at the design point is added to the system pump curve, specifically to offset any long-term degradation of pump performance during the life of the plant.

There is also a SR in the SWS investment protection short-term availability controls to verify that each SWS pump provides a flow rate of 37,850 Lpm (10,000 gpm) one day prior to entering Mode 5 and reduced-inventory operation.

The staff's review of the applicant's response to RAI-SRP-9.2.1-SBPA-01 finds that it adequately explains how the margins in SWS system design were developed. Also the staff finds that the response adequately explained how the flow margins would be tested and maintained for the life of the plant since testing/surveillance is performed before entering Mode 5. The staff also reviewed the proposed changes to DCD Section 9.2.1.2.3.4 and finds that they adequately provide a high level description of the SWS margin. Therefore, RAI-SRP9.2.1-SBPA-01 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Design Considerations

In addition to accommodating the higher CCS heat load while maintaining appropriate SWS flow velocities and pressure drops, the SWS return temperature to the cooling tower is not allowed to exceed 48.9 °C (120 °F) under design heat load conditions. As discussed in Section 3.4.1 of TR-111, this limit is set to prevent long-term degradation of the cooling tower fill material. In order to assure that this temperature limit will not be exceeded upon initiation of shutdown cooling when the SWS heat load is maximized at 3.65 x 10^8 kJoules/hr (3.46 x 10^8 Btu/hr), the applicant proposes to use larger capacity SWS pumps with a flow rate of 37,747 Lpm (10,500 gpm) per train. The use of larger capacity pumps will avoid delays when cooling down the plant in preparation for refueling and this is considered to be acceptable by the staff since the shutdown timeframe is enhanced. However, this design capability does not ensure that plant operators will adhere to this temperature limit. The staff requested, in RAI-SRP9.2.1-SBPA-02, that the applicant identify and describe in Tier 2 of the DCD those SWS design limitations that should be adhered to and explain how adherence to these limitations is assured. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant provided additional information about the system design margins, alarms, and capabilities, and discussed cooling tower fill material options that are available for higher temperature situations. However, the response did not adequately address the staff's request in that SWS design limitations and provisions to ensure adherence were not discussed. More information was requested in RAI-SRP9.2.1-SBPA-02 regarding RTNSS and systems supporting the ability to achieve cold shutdown operations.

During the June 25, 2009 audit, the staff explained that in order to resolve this RAI, the applicant should provide more detail identifying in Tier 2 of the DCD those SWS design limitations that should be adhered to and to explain how these limitations are assured including the addition of instrumentation if required. In a letter dated August 31, 2009, the applicant provided additional information specific to the design limitations appended to its original response to RAI-SRP9.2.1-SBPA-02.

The applicant agreed that SWS cooling tower basin water level Instrumentation was necessary and should be included in ITAAC. The response stated that this parameter is needed to verify that SWS pumps will be supplied with adequate net positive suction head (NPSH). Testing to verify adequate NPSH is also discussed in Tier 2 Section 14.2.9.2.6, "Service Water System Testing," in the item (d) of the 'General Test Acceptance Criteria and Methods. The applicant proposed to revise DCD Tier 1 Table 2.3.8-2 to identify the service water cooling tower basin level instrument, SWS-009.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.1-SBPA-02 and finds that it provides an adequate means to measure the service water cooling tower basin level and ensures adequate SWS NPSH that can be verified by testing. The staff also reviewed the proposed changes to the DCD and finds that they adequately provide the instrumentation required to ensure adequate SWS NPSH. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The proposed increase in SWS capacity and flow rate is expected to result in an increase in the minimum water inventory that must be maintained in the cooling tower basin. The applicant did not describe specifically what water inventory should be maintained in the cooling tower basin in order to support SWS operation and, in particular, to assure adequate net positive suction head for the SWS pumps. In RAI-SRP9.2.1-SBPA-03 the staff asked the applicant to identify and describe in Tier 2 of the DCD the cooling tower basin water inventory requirements, the basis for this determination, and how this inventory is assured to be maintained. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant provided information to explain the design capability of the cooling tower basin inventory and while this information is useful, specific quantitative information is needed to verify the adequacy of design. For example, the specific NPSH requirements for the SWS pumps was not provided and design details of the cooling tower basin were not provided to show that the NPSH requirement was satisfied; other considerations such as vortexing were not addressed. Furthermore, this information was not included in the Tier 1 and Tier 2 descriptions as appropriate, and there was no discussion about how implementation of operational assumptions is assured. Additionally, the response made reference to use of a raw water system (RWS) for providing makeup to the SWS cooling tower basin, which has not been described for the AP1000 design in accordance with 10 CFR 52.47(24), "Contents of applications; technical information" requirements. Consequently, the response did not adequately address RAI-SRP9.2.1-SBPA-03.

During the June 25, 2009 audit, the staff explained that in order to resolve this RAI, the applicant should provide more information on its evaluation of whether it is reasonable to have a cooling tower level requirement prior to going to mid-loop operations for RTNSS considerations, and entering Modes 5 and 6. The staff also asked the applicant to evaluate and describe the differences in water level versus usable volume (in the SWS cooling tower basins) between normal operation and RTNSS.

In a letter dated August 31, 2009, the applicant provided additional information specific to the design limitations appended to its original response to RAI-SRP9.2.1-SBPA-03. In the response, the applicant agrees with the need to add SWS cooling tower basin usable volume to the ITAAC for SWS. A minimum SWS cooling tower basin reserve volume is required to provide water inventory for up to 12 hours when normal makeup capability from the RWS is lost. The basin is sized to contain 870,600 L (230,000 gallons) between the low level alarm setpoint (elevation 30.2 m (99 ft)) and the lowest usable level (28.5 m(93 ft, 6 in)), which coincides with the low-low level alarm setpoint. This is the minimum cooling tower usable volume needed to support a plant cooldown for 12 hours without makeup. The response states that a criterion will be added to the SWS ITAAC to ensure that the SWS cooling tower basin is constructed to provide this minimum reserve volume with water level at the low level alarm setpoint. Additionally, the applicant proposes to revise Table 2.3.8-2, ITAAC Item 2 to add testing requirements confirming that the SWS cooling tower basin has adequate reserve volume of at least 870,600 L (230,000 gallons) corresponding to its low level alarm setpoint. Additionally, the applicant proposed to revise Tier 2 DCD Section 9.2.1.2.2 for the cooling tower to identify that a minimum usable volume of 870,600 L (230,000 gallons) exists.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.1-SBPA-03 and finds that it adequately identifies that the service water cooling tower basin level will have a usable minimum of 870,600 L (230,000 gallons) corresponding to its low level alarm setpoint. The staff also reviewed the proposed changes to the DCD and finds that they adequately provide ITAAC criteria ensuring the testing to confirm the usable volume. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The increased water inventory and flow rates in the SWS piping and cooling towers could result in more severe flooding consequences than previously analyzed. The information that was provided did not describe the impact of the proposed SWS modifications on the consequences of flooding and whether or not safety-related equipment could be adversely affected. The applicant was asked, in RAI-SRP9.2.1-SBPA-04, to address the potential impact of the proposed SWS modifications on safety-related equipment and on the consequences of flooding.

In a letter dated June 26, 2008, the applicant provided information specific to flooding. In the response the applicant stated that there is no safety-related equipment in the turbine building. The component cooling water and service water components on elevation 30.5 m (100 ft), which provide the RTNSS support for the normal RNS are expected to remain functional following a flooding event in the turbine building since the pump motors and valve operators are above the expected flood level. Flooding caused by an SWS piping break in the compartment at the southern end of the turbine building will flow out the open doorway to the turbine hall, and openings in the base of the wall between the compartment and the turbine hall. The increased flooding rate and volume associated with the increase in SWS pump size and basin inventory will not challenge the operability of the CCS pumps nor any other DID or Investment Protection equipment located in this area. Flooding from the circulating water system (CWS) (or from any other turbine hall water source) will flow out of the turbine building onto the ground through access doors without affecting equipment at the southern end of the building.

The staff reviewed the RAI response and concluded that the RTNSS components (CCW and SWS) would remain functional since the pump motors and valve operators are above the expected flood levels and there are no safety-related components in the turbine building. Therefore, RAI-SRP9.2.1-SBPA-04 is considered closed.

9.2.1.2.2 Proposed Increase in the Maximum SWS Supply Temperature

Proposed Revision 16 of the DCD, Tier 2, Sections 9.2.1.1.2 and 9.2.1.2.3.3, reflects an increase in the maximum allowed SWS cooling water temperature being supplied to the CCS HXs during normal power operation. The proposed temperature increase is from a value of 31.7 °C (89 °F) to 34.2 °C (93.5 °F). This is not the same as the maximum SWS supply temperature of 31.4 °C (88.5 °F) that is specified in Tier 2 of the DCD, Section 9.2.1.2.3.4, for plant cooldown/shutdown. This change is not discussed or explained in the information that the applicant provided, and it is not clear how the SWS can perform its DID and RTNSS functions if the SWS supply temperature exceeds the limit that is assumed for shutdown cooling. In RAI-SRP9.2.1-SBPA-05 the staff requested that the applicant describe in Tier 2 of the DCD the basis and justification for the proposed increase in the maximum allowed normal operating SWS supply temperature.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant discussed differences in assumptions that were used for dissipating heat during normal power operation and during plant cooldown. The difference in the cooling water temperature values noted above stems from the use of a less conservative wet bulb temperature for the most limiting SWS DID function (plant cooldown with concurrent spent fuel cooling). However, the applicant did not explain how the SWS DID and RTNSS functional capabilities are assured for those periods when humidity is at its maximum. There is an increased chance of plant trip during hot, humid conditions due to increased electrical demand and it is illogical to assume less humid conditions for the plant cooldown case than what can be experienced during normal power operation. In accordance with SECY-94-084, the SWS should be capable of performing its DID and RTNSS functions over the full range of postulated operating conditions, and the applicant has not demonstrated this to be the case. The applicant response also referred to another increase that is proposed for the maximum safety wet bulb temperature in order to accommodate the Levy site. Because the RAI response did not adequately address the DID and RTNSS functional capabilities of the SWS over the full range of plant operating conditions, satisfactory resolution of RAI-SRP9.2.1-SBPA-05 was not achieved.

On March 18, 2009, the staff conducted a public meeting with the applicant to discuss this RAI. After this meeting, the applicant provided a revised RAI response on May 13, 2009, which addressed wet bulb temperature values related to the SWS to accommodate the Levy site environmental parameters. The applicant stated that the maximum normal non-coincident wet bulb temperature remains at 26.7 °C (80.1 °F) for RTNSS and DID SWS functions and that there was sufficient margin in the system and component design.

In a letter dated May 13, 2009, the applicant stated that higher ambient temperatures (30.0 °C (86.1 °F) vs. 26.7 °C (80.1 °F)) will not impact safety or investment protection and would only result in an extended time to achieve cooldown. During the June 25, 2009 audit, the applicant clarified its original response and supplemental responses. The staff agreed that the responses were acceptable because safety was not impacted while cooldown time was extended; therefore, RAI-SRP9.2.1-SBPA-05 is resolved.

9.2.1.2.3 Impact of Revised Site Interface Temperature Limits on Cooling Tower Performance

As discussed in TR-108, the applicant proposes to change the site interface temperature limits to encompass a broader range of potential sites for AP1000 plants. In particular, Tier 1 of the DCD, Table 5.0-1, “Site Parameters,” is revised to specify a maximum (noncoincident) wet bulb temperature of 29.7 °C (85.5 °F) instead of 27.2 °C (81 °F). Tier 2 of the DCD, Table 2-1, “Site Parameters,” is revised to reflect this higher (noncoincident) wet bulb temperature as the Maximum Safety (or 0 percent exceedance) value, and the Maximum Normal (or 1 percent exceedance) wet bulb temperature is revised from 25 °C (77 °F) to 26.7 °C (80.1 °F) for the coincident value and from 26.7 °C (80 °F) to 26.7 °C (80.1 °F) for the noncoincident value. The proposed change to the maximum normal noncoincident value is reflected in the DCD Tier 1 ITAAC that are specified in Table 2.3.8-2 and is referred to in DCD Tier 2, Section 9.2.1.2.3.4, “Plant Cooldown/ Shutdown.”

The ITAAC requires COL applicants to demonstrate that each cooling tower cell is capable of dissipating the specified shutdown and spent fuel heat loads at the maximum normal (non-coincident) wet bulb temperature. The proposed change in the ITAAC relative to the wet bulb temperature assumption is consistent with the proposed changes in the site interface temperature limits, and it is acceptable from this perspective. However, the difference between the maximum normal and maximum safety non-coincident wet bulb temperatures was rather trivial, only 0.5 °C (1 °F) before the proposed change; but after the proposed change the gap is widened to 3 °C (5.4 °F). This larger delta between the maximum normal and maximum safety non-coincident wet bulb temperature values warrants further consideration to assure that cooling tower performance is adequate for accomplishing its DID and RTNSS functions. Also, because Tier 2 of the DCD, Sections 5.4.7.1.2.3 and 9.2.2.1.2.1 indicate that cooling tower performance is based upon the maximum safety non-coincident wet bulb temperature as a limiting assumption, it is not clear how this capability is assured. Furthermore, the cooling tower cold water temperature that is specified by the ITAAC is revised from 37.8 °C (100 °F) to 32.2 °C (90 °F) without explanation or justification. This does not appear to be consistent with the supply temperature of 34.2 °C (93.5 °F) that is assumed in Tier 2 of the DCD, Section 9.2.1.1.2. Given these observations, the staff requested In RAI-SRP9.2.1-SBPA-06 that the applicant identify and explain in Tier 2 of the DCD the limiting assumptions and bounding conditions that are important relative to cooling tower design, performance, and operation for assuring that the SWS is capable of and can be relied upon to perform its DID and RTNSS functions, what provisions exist to ensure that these limiting assumptions and bounding conditions will be satisfied by COL applicants over the life of the plant, and what the potential consequences are of exceeding the maximum normal non-coincident wet bulb temperature during operating and shutdown conditions.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant explained that the DID functions for the SWS are based on the maximum normal non-coincident wet bulb temperature for the site. The response also indicated that the SWS cooling function is not needed for maintaining the plant in a long-term safe condition. The applicant further indicated that SWS flow and cooling tower performance would be monitored on a continuous basis and that licensees will perform testing at regular intervals to determine cooling tower heat transfer capability, but a COL action item does not exist and one was not established to specify this action by COL applicants. The RAI response did not adequately address RAI-SRP9.2.1-SBPA-06 because more information was needed regarding the DID and RTNSS functional capability of the SWS over the full range of plant operating conditions.

During the June 25, 2009 audit, the staff expressed a concern regarding system reliability for RTNSS and DID over full range of operating conditions. The staff referred to similarities to RAI-SRP9.2.1 SBPA-01 and RAI-SRP9.2.1 SBPA-02 above, discussing what instruments are available to the control room for monitoring system performance.

The staff determined that the revised response to RAI-SRP9.2.1-SBPA-01 above resolves this issue because adequate explanation was provided with respect to RTNSS, cooldown, and DID considerations. Further, the discussion identified that the answer to the instrumentation portion of this concern can be found as part of the response to RAI-SRP9.2.1-SBPA-02. Therefore, RAI-SRP9.2.1-SBPA-06 is resolved.

In APP-GW-GLE-036 , the applicant described the impact of changing the current Maximum Safety wet bulb non-coincident temperature from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F) and the maximum safety wet bulb coincident temperature from 26.7 °C (80 °F) to 30.1 °C (86.1 °F) to encompass more sites in the eastern United States. In APP-GW-GLE-036, the applicant performed an evaluation of the effect on CCS and SWS to determine if sufficient margin exists to accommodate these changes in Maximum Safety wet bulb temperatures.

The CCS and SWS system provide heat removal from numerous plant loads for normal and abnormal modes of operation. During power operation, the systems are designed to accommodate the Maximum Safety temperature conditions (0 percent exceedance) with a single train in service whereas during shutdown operations the system is designed for the Maximum Normal conditions (1 percent exceedance) with both trains in service.

The applicant concluded that the SWS cooling tower is not expected to require changes to accommodate the higher Maximum Safety wet bulb temperature since the cooling tower sizing case is for plant cooldown at 4 hours after reactor shutdown, and is based on the Maximum Normal 1 percent exceedance value of 26.7 °C (80.1 °F), which is unchanged. The SWS cooling water supply temperature for the Maximum Safety case will be 33.1 °C (91.6 °F), resulting in a maximum CCS supply temperature of 36.1 °C (97.0 °F).

Based on its review of the changes to maximum safety wet bulb temperatures as evaluated by the applicant in APP-GW-GLE-036, the staff finds that the applicant adequately explained the increase in wet bulb temperature limits identified in DCD Tier 1 Table 5.0-1 and Tier 2 Table 2-1 and the applicant demonstrated that the changes in wet bulb temperature limits would not adversely impact performance of the CCS.

9.2.1.2.4 Miscellaneous Changes

DCD Tier 2, Revision 17, incorporates two changes in Section 9.2.1 that are described in TR-103. These changes include provisions for using nonmetallic piping and the removal of "smart valves."

Use of Nonmetallic Piping

The applicant proposed a change to Tier 2 of the DCD, Section 9.2.1.2.2, "Component Description," to allow COL applicants the option of using black polyethylene piping (High Density Polyethylene or HDPE) for SWS applications in accordance with the ASME B31.1 Power Piping Code if deemed appropriate by evaluation. In particular, HDPE could be used in areas of low pressure and low temperature, up to 1,034 kilopascal (kPa) (150 psi) and 60 °C (140 °F). Although the SWS is subject to RTNSS, it is not relied upon for post-72 hour cooling following an accident and the design provisions that pertain to seismic, flooding, and hurricane conditions do not apply. Therefore, from this perspective, the proposed use of HDPE is acceptable. However, since the SWS function is considered to be risk important during shutdown conditions when the reactor is open, the impact of using HDPE on SWS reliability and availability assumptions should be considered and addressed. Also, the review criteria specified by NUREG-0800 Section 3.6.1 relative to pipe failure evaluations is based on the use of metal pipe. Unless otherwise justified by the applicant, the potential consequences of pipe failure (including flooding) should be evaluated assuming the complete failure of all HDPE piping during seismic events coincident with metallic pipe failures that are postulated and other considerations that are specified by NUREG-0800. Finally, the specific criteria for allowing the use of HDPE should be specified in the DCD to ensure clarity of the plant licensing basis. The applicant was asked, in RAI-SRP9.2.1-SBPA-07, to revise the DCD (Tier 1 and Tier 2 as appropriate) to address these considerations.

The applicant responded to the staff's request in a letter dated June 26, 2008, and referred to its earlier response to RAI-TR103-EMB2-02 dated February 22, 2008. Also, additional clarifying information was provided to specify that HDPE will be used for the underground portions of the auxiliary makeup line from the secondary fire water tank and for the underground portions of the SWS blowdown to the circulating water system cooling tower. However, the applicant did not address the specific question that was asked by the staff in RAI-SRP9.2.1-SBPA-07. As a separate matter, the staff also requested the applicant to describe how the requirements specified by 10 CFR 20.1406, "Minimization of Contamination" are satisfied with respect to SWS considerations, including provisions that have been established for buried SWS pipe. Consequently, the RAI response did not adequately address RAI-SRP9.2.1-SBPA-07 because more information was needed regarding the use of nonmetallic piping and its potential effects on RTNSS and systems supporting the ability to achieve cold shutdown operations.

In a letter dated August 31, 2009, the applicant provided additional information specific to the use of HDPE nonmetallic piping in the SWS by proposing to revise DCD Tier 2 Section 9.2.1.2.2, "Piping," to specify that instead of nonmetallic piping, only high density polyethylene piping is used for the underground portions of the auxiliary makeup line from the Secondary Fire Water tank, and for the underground portions of the SWS blowdown line to the CWS cooling tower.

The staff reviewed the applicant's additional information in its response to RAI-SRP-9.2.1-SBPA-07 and finds that it adequately explains the use of nonmetallic piping in the SWS system design. The staff also reviewed the proposed changes to DCD Section 9.2.1.2.2 and finds that they adequately specify that only HDPE is used in the SWS. Therefore, RAI-SRP9.2.1-SBPA-07 was resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Related to 10 CFR 20.1406, in the applicant's submittal of TR-98, "Compliance with 10 CFR 20.1406," APP-GW-GLN-098, dated April 10, 2007, and the applicant's response to RAI-SRP- 12.1-CHPB-01, dated September 9, 2008, all radioactive piping is located inside the auxiliary building which minimizes the potential for leakage to the groundwater from piping and fittings. In addition, no piping containing radioactive fluid is directly buried in the ground. Based on the staff's review, the staff determined 10 CFR 20.1406 has been adequately addressed since the buried SWS does not normally contain radioactive fluids. In the event the SWS contains radioactive fluid, a radiation monitor with a high alarm is provided to monitor the service water blowdown flow. The component cooling water HXs and tower blowdown flow can be isolated by remote manual control. 10 CFR 20.1406 design considerations are further discussed in Chapter 12 of this report.

Removal of Smart Valves

The AP1000 design specifies the use of "smart valves" (i.e., valves that contain instrumentation such as temperature, flow and pressure that is used for control or indication) for some system applications. In the case of the SWS, smart valves (V009 and V011) are specified for the cooling tower makeup and blowdown control valves. The applicant proposed to remove the requirement for using smart valves for these functions to provide flexibility in the design for COL applicants. The proposed change replaces the instrumentation that is included in the smart valve design with standard inline instrumentation as illustrated in Figure 9.2.1-1 of the DCD, Tier 2, and references to valves with internal instrumentation are removed from the description provided in Section 9.2.1.5. This proposed change does not eliminate or alter the functional capabilities of any SWS valves or instruments, and should not degrade the capability or reliability of the SWS to perform its function. If anything, this less complicated arrangement is expected to improve the capability to service and maintain the affected instrumentation, which would tend to improve SWS availability and reliability consistent with SECY-94-084. Therefore, the staff considers the proposed changes to use standard inline instrumentation to be acceptable.

9.2.1.2.5 Investment Protection Short-Term Availability Controls (IPSAC)

The applicant proposed to change the minimum required flow rate specified by SR 2.4.1 in DCD Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," to be consistent with the SWS design changes discussed above in Section 9.2.1.1.1. The minimum required flow rate for each SWS pump would be changed from more than 32,555 Lpm (8600 gpm) to more than 37,854 Lpm (10,000 gpm). The proposed change is consistent with the minimum required SWS flow rate discussed in Section 9.2.1.1.1 and from this perspective, it is appropriate. On this basis, the staff finds this acceptable.

9.2.1.2.6 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Considerations

The applicant proposed to change SWS flow rate, heat dissipation capability, and temperature conditions that are specified in ITAAC Table 2.3.8-2 to reflect proposed changes that are discussed above in Sections 9.2.1.1.1 and 9.2.1.1.3. The changes that are proposed for the SWS ITAAC are consistent with the specific changes that are proposed in these sections. On this basis the staff finds this acceptable.

9.2.1.2.7 Initial Test Program Considerations

The initial test program for the SWS is discussed in Tier 2 of the DCD, Section 14.2.9.2.6, "Service Water System Testing." The stated purpose of the SWS test program is to verify the capability of the as-installed system to transfer heat from the CCS HXs to the environment. Prerequisites are specified to assure that the SWS is properly configured and ready for testing, that plant conditions are appropriate, and provisions for collecting data have been established as necessary. SWS performance is observed and recorded during a series of individual component and integrated system tests in order to demonstrate that the SWS properly performs its DID functions. While most of the testing appeared to be appropriate and adequate for demonstrating the SWS DID capabilities, the staff noted that adequate performance for the most limiting situations was not specifically addressed, such as confirming adequate NPSH for the most limiting level, temperature, and flow rate situations and likewise for confirming adequate cooling tower performance; and adequate cooling tower makeup capability does not appear to be verified by the testing that is completed.

In RAI-SRP9.2.1-SBPA-08, the applicant was asked to address these considerations and in particular, to explain how the DID capability of the SWS is assured for the most limiting situations such that system reliability and availability assumptions are valid. The applicant responded to the staff's request in a letter dated June 26, 2008, referring the staff to Tier 2 Section 14.2.9.2.6 of the DCD. The applicant also provided additional information concerning the nature and extent of testing that would be performed. However, much of the information was not reflected in Tier 2 of the DCD, and the stated purpose of the test continues to be very narrowly focused on demonstrating the capability of the CCW HXs to transfer heat to the environment. For example, testing is not specified to confirm that hydraulic transients will not occur; especially following a loss of power and potential drain down of the system. Testing is not specified to demonstrate satisfactory performance during limiting conditions; the cooling tower makeup capability is not specified and confirmed, etc. Furthermore, the applicant credits COL applicants for performing ongoing surveillance and testing, but there are no COL action items to this effect.

In a letter dated August 31, 2009, the applicant provided additional information specific to the design margins, appended to its original response to RAI-SRP-SBPA-08 to clarify that Section 9.2.1.2.3.6 in Tier 2 of the DCD, actions are taken to prevent drain down and water hammer in the SWS during a loss of normal alternating current (ac) power. The motor-operated SWS tower inlet valves, which are loaded onto the diesel generators (DGs), are automatically closed when power is lost.

The applicant further identified that the diesel-backed SWS pumps undergo their normal automatic start procedure, which is described in detail in Section 9.2.1.2.2 in the last paragraph of the 'Service Water Pumps' section. Since the motor-operated SWS pump discharge valve and the tower inlet valve are both powered by the DG, these valves can stroke open to allow partial flow during pump start, thereby maintaining a water solid system.

Further the applicant stated that Tier 2 Section 14.2.9.2.6 includes testing for the instrumentation, controls, actuation signals and interlocks, as described in item (b) of the 'General Test Acceptance Criteria and Methods' section. As indicated in the first bullet, automatic pump actuation is verified in case an operating pump stops. The testing of the valve interlocks will ensure that they are able to perform their automatic start functions during a loss of power transient.

Service water blowdown is also isolated during a loss of normal AC power condition, in order to reduce liquid loss from the system. The SWS blowdown flow control valve is designed to automatically close when power is lost, which is an interlock that will also be tested in accordance with Section 14.2.9.2.6. This flow control valve is isolated using an ac-powered solenoid, which is fed from a protected, inverter-backed bus.

The staff reviewed the applicant's additional information in its response to RAI-SRP-9.2.1-SBPA-01 and RAI-SRP-9.2.1-SBPA-08 and finds that it adequately explains SWS system operation during normal operation, testing, and operation during loss of normal ac power. Therefore RAI-SRP9.2.1-SBPA-08 is resolved.

9.2.1.3 Conclusions

The staff evaluated proposed changes to Revision 15 of the AP1000 DCD that pertain to the SWS. The proposed changes are documented in Revision 17 of the AP1000 DCD, Tier 2, Section 9.2.1, and are reflected in the Tier 1 ITAAC specified in Table 2.3.8-2 and the Tier 2 IPSAC specified in DCD Table 16.3-2, Section 2.4. The staff's evaluation, using the guidance provided by NUREG-0800 Section 9.2.1, confirmed that: a) the proposed changes will not adversely affect safety-related SSCs, b) the SWS is capable of performing its DID and RTNSS functions, and c) ITAAC, IPSAC, and initial test program considerations are adequate and appropriate. The proposed changes conform to the existing AP1000 licensing basis as documented in Revision 15 of the approved DCD. The proposed changes meet the criteria of; 10 CFR 52.63(a)(1)(iii), on the basis that they reduce unnecessary regulatory burden while maintaining protection to public health and safety and the common defense and security; and 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information in the AP1000 DCD.

9.2.2 Component Cooling Water System

9.2.2.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.2.2, “Component Cooling Water System,” of the AP1000 DCD, Revision 15. In the AP1000, Revision 17, the applicant proposed changes to Section 9.2.1.

The applicant proposed the following technical changes to Revision 15 of the AP1000 DCD, which are supported by information in the TRs:

1. Proposed changes to the CCS in order to accommodate increased heat loads. This includes the CCS pump design capacity which was changed from 33900 Lpm (8960 gpm) to 35900 Lpm (9500 gpm).

2. The applicant changed the maximum CCS supply temperature to plant components from 35 °C (95 °F) to 37.8 °C (100 °F). Additionally, the applicant raised the wet bulb temperature for service water cooling at normal operations (maximum normal temperature per Table 2-1 for normal shutdown). The ambient design wet bulb temperature was raised from 29.7 °C (85.5 °F) to 30.1 °C (86.1 °F).

3. The applicant changed the RCPs component cooling water discharge isolation valves from MOV to air operated valves (AOV). The applicant changed the closing signal input to these valves to an AOV close signal generated by the plant control system. The close signal will use simultaneous CCS flow deviations in the RCP’s CCS supply and return lines. The applicant deleted reference to a tube rupture in the RCP motor cooling coil or thermal barrier as a source of reactor coolant leakage into the CCS and replaced this source of potential leakage with the RCP external HX. These changes conform to the applicant’s RCPs design change to canned RCPs, which includes a new RCP external HX that replaces the RCP motor cooling coils and thermal barriers what were cooled by CCS.

4. The applicant described changes to the flow sensors in the CCS inlet and outlet lines associated with each RCP external HX. The proposed change clarified that isolation valves used to isolate the component cooling water outlet line associated with its RCP are nonsafety-related. The changes associated with a canned RCP include the installation of a RCP external HX to replace the pump motor cooling coil and thermal barrier. The applicant identified that other plant alarms and indications can be used by the operator to manually initiate a RCP component cooling water system isolation.

5. The applicant provided additional detail for instrumentation for high-level and low-level alarms on the CCS surge tank, automatic actuation of the CCS surge tank makeup water valve for makeup flow from the demineralized water transfer and storage system into the CCS, and flow alarms in the MCR to indicate that a leak exists on the RCP external HX. The RCP external HX replaces the RCP motor cooling coils and thermal barriers and is cooled by CCS. In addition, the applicant identified that flow measuring instrumentation on the RCP component cooling water inlet and outlet lines provide an isolation signal to close an AOV and isolate the affected RCP external HX from the rest of the CCS.

6. The applicant corrected the referenced code to use for nonmetallic piping from "ASME B31.1" to "ANSI B31.1." The proposed change also clarified that ANSI B31.1 Appendix III also may be used for outside containment piping.

Details of these proposed changes are discussed in TR-111, TR-108 and TR-103, respectively.

9.2.2.2 Evaluation

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal DID capability of removing reactor and spent fuel decay heat. It is also part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with SECY-94-084. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its DID and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was judged based upon conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.2 (as applicable), and SECY-94-084.

Modifications to approved standard plant DCs can be proposed provided (among other things) that the changes are deemed to be necessary in accordance with 10 CFR 52.63(a)(1). The proposed changes will allow additional flexibility for COL applicants, thereby reducing the need for departure requests. The specific criteria that apply to the proposed changes referred to above include: 10 CFR 52.63(a)(1)(iii), which concerns reducing unnecessary regulatory burden while maintaining protection to public health and safety and the common defense and security; and 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

During its evaluation, the staff noted that the description that is provided in Tier 2 of the AP1000 DCD, Section 9.2.2, "Component Cooling Water System," does not describe the DID, investment protection, and RTNSS design basis for the CCS. However, it is clear from the ITAAC specified in Tier 1 of the DCD, Section 2.3.1, "Component Cooling Water System;"," the initial test program described in Tier 2 of the DCD, Section 14.2.9.2.5, "Component Cooling Water System Testing," Table 16.3-2, "Investment Protection Short-Term Availability Controls," as it pertains to CCS, and Table 17.4-1, "Risk-Significant SSCs Within the Scope of D-RAP," that the CCS is important for both DID, investment protection, and RTNSS considerations. However, this information has not been adequately reflected in the description that is provided for the CCS in Tier 2 of the DCD, Section 9.2.2. In RAI-SRP9.2.2-SBPA-03 the applicant was asked to include additional information in Section 9.2.2 to better explain the DID, investment protection, and RTNSS design basis for the CCS (also see test considerations referred to below in Section 9.2.2.2.7). The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant indicated that the information provided for the CCS is similar to what was provided for the SWS and other DID systems. The staff confirmed that the Tier 2 information for the CCS is similar to what was provided for other systems of this nature, with no description of the DID, investment protection, or RTNSS design basis for these systems.

During the June 25, 2009 audit, the staff explained that, in order to resolve the RAI, the applicant should provide more detail regarding the CCS RTNSS, Design Reliability Assurance Program (D-RAP), and IPSAC functions. In a letter dated August 31, 2009, the applicant provided additional information about the CCS functions by proposing to revise DCD Tier 2 Section 9.2.2 referring to DCD Tier 2 Section 17.4-1.

The staff reviewed the applicant's additional information in its response to RAI-SRP-9.2.2-SBPA-03 and finds that it adequately explains the CCS D-RAP functions by referencing DCD Tier 2 Section 17.4. The staff also reviewed the proposed changes to DCD Section 9.2.2.3.1 and finds that they adequately referenced the basis for including the CCS components within the scope of D-RAP. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.2.2.2.1 CCS Design Changes Required to Accommodate Increased Heat Loads

As discussed in TR-111, CCS modifications are necessary as a result of higher heat loads and flow rates for cooling the RCPs and the condensate pump oil coolers, adding the VFD for the RCPs as new CCS heat loads and relocating them from the northwest to the southwest side of the turbine building, and excessive CCS flow velocities at the CCS pump suction and discharge headers and inside containment due to the increased CCS flow demand that is necessary to satisfy increased component heat loads. Consequently, the applicant implemented CCS design changes to add new components, reconfigure the CCS piping layout and resize pipe as necessary, revise CCS pump and HX parameters, and increase the CCS design pressure. The CCS changes are reflected in the AP1000 DCD, Tier 1, Table 2.3.1-2, ITAAC Design Commitment, Item 3, relative to the capability to support plant shutdown and spent fuel cooling; Tier 2, Section 9.2.2, “Component Cooling Water System,” including Table 9.2.2-1, “Nominal Component Data – Component Cooling Water System,” Table 16.3-2, “Investment Protection Short-Term Availability Controls,” Section 2.3, “ Component Cooling Water System (CCS) – RCS Open,” SR 2.3.1 with respect to the minimum required CCS flow rate.

Relocating the VFDs to the southern end of the turbine building places them in close proximity to the CCS pumps and HXs. Failures associated with the VFDs could affect the capability of the CCS to perform its RTNSS function and additional information is needed to address this consideration. In RAI-SRP9.2.2-SBPA-04, additional information was requested to address relocating the VFDs. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant stated that typical failures expected for high power electronic equipment include fires and in this case loss of cooling water from the dedicated cooling system or from the CCS which supplies cooling water to the VFD internal cooling system HXs. Fires in the turbine building, caused specifically by a failure of VFD equipment, that disable both CCS pumps have been addressed by the inclusion of a means to provide 2271 Lpm (600 gpm) of cooling water to normal RNS HX ‘A’ from the FPS to provide continued capability to remove decay heat from the RCS following suppression of the fire. During suppression activities, the plant passive safety systems ensure that decay heat is removed from the core and therefore cooling of the RNS HXs with CCS is not required. SFS pool cooling is also provided by other means during this period of time. These provisions are described in DCD Revision 16, Tier 2, Sections 9.1.3.4.3, “Abnormal Conditions,” and 9.2.2.4.5.5. In addition, a break in the VFD internal cooling water lines or in the CCS lines supplying the HXs does not increase the risk of a flooding event, as a break of this size is enveloped by the bounding flooding case of breaks in larger CCS and SWS lines in the southern end of the turbine building.

The staff reviewed this response and determined that it did not address RTNSS considerations. In a revised RAI response dated April 13, 2009, the applicant stated that the variable frequency drive is used only during heatup and cooldown when the reactor trip breakers are open. During power operations, the drive is isolated and the RCP is run at constant speed; therefore, the VFDs are de-energized during suppression activities.

The staff reviewed the revised RAI response and determined that RTNSS considerations were adequately addressed since the flooding event is bounded by the larger CCS and SWS line breaks in the turbine building and the VFDs would not be energized during the period of time the CCS would be performing their RTNSS function in Mode 5 and 6. Therefore, RAI-SRP9.2.2-SBPA-04 is considered resolved.

In DCD Revision 16, the applicant reanalyzed the fluid pressures throughout the redesigned CCS and determined that the design pressure of the system should be increased from 1,034 kPa (150 pounds per square inch gauge (psig)) to 1,379 kPa (200 psig). However, the total design differential head of the CCS pumps is actually reduced substantially and it was not clear why the system pressure was increased. The staff asked the applicant, in RAI-SRP9.2.2-SBPA-05, to address this inconsistency. In response to RAI-SRP9.2.2-SBPA-05, the applicant referred to industry operating experience showing that relief valve actuations occur frequently during routine realignments (e.g., pump swaps) in CCS that are designed for 1,034 kPa (150 psig). Also, based upon the results of a hydraulic analysis that was performed, the applicant determined that the CCS operating pressure for AP1000 is just below the relief valve setpoint for a system design pressure of 1,034 kPa (150 psig). Consequently, the applicant increased the CCS design pressure to 1,379 kPa (200 psig) in order to minimize the occurrence of relief valve actuations and the likelihood of valve leakage. Based on the information that was provided, the staff finds that the proposed increase in the CCS design pressure will increase the available margin and make the system more robust. Additionally, because the higher relief valve set point will tend to minimize spurious actuations and valve leakage, the proposed increase in CCS design pressure reduces the likelihood of spreading radioactive contamination consistent with 10 CFR 20.1406 requirements. Therefore, the staff considers the proposed increase in CCS design pressure to be acceptable and RAI-SRP9.2.2-SBPA-05 was resolved.

Proposed Increases in CCS Flow Rate and Heat Removal Capability

The applicant proposed to increase the minimum required CCS flow rate and heat removal capability in order to accommodate the design changes referred to above. The values listed in Tier 2 of the DCD, Section 9.2.2 and Table 9.2.2-1, and the ITAAC specified by Tier 1 of the DCD, Table 2.3.1-2, reflect these changes.

The ITAAC specified in Tier 1 of the DCD, Table 2.3.1-2, require COL applicants to demonstrate that the CCS design is capable of supporting plant shutdown and spent fuel cooling. The applicant proposes to change the ITAAC acceptance criteria to demonstrate a flow rate for each CCS pump of at least 10,164 Lpm (2685 gpm) to one normal shutdown cooling HX (this is unchanged), plus 4542 Lpm (1200 gpm) to one SFP HX (this is increased by 284 Lpm (75 gpm) from the previous amount), and at least 16,713 Lpm (4415 gpm) to other CCS heat loads (this is increased by 12,397 Lpm (3275 gpm) from the previous amount), for a total required flow rate for each CCS pump of 31,419 Lpm (8300 gpm). This represents an increase in the total required flow rate for each CCS pump of 12,681 Lpm (3350 gpm). The total CCS pump flow rate that is specified for each pump is consistent with the proposed value that is listed in Tier 2 Table 9.2.2-1, and it is acceptable from this perspective. However, Tier 2 of the DCD, Section 9.2.2, does not identify what the minimum CCS flow requirements are for these three categories of heat loads that are listed in the ITAAC, how much excess margin is available for each one, the basis for this determination, and how the specified flow balance will be maintained over time. In RAI-SRP9.2.2-SBPA-06, the staff requested that the applicant address this missing information.

The applicant responded to RAI-SRP9.2.2-SBPA-06 in a letter dated June 26, 2008. The applicant provided additional information primarily related to CCS HX design and made reference to TR-111 for additional discussion. However, the applicant's response did not address the specific question that was asked by the staff.

During the June 25, 2009 audit, the staff explained that, in order to resolve the RAI, the applicant should provide more detail concerning how the ITAAC demonstrate adequate flow for RTNSS.

In a letter dated August 31, 2009, the applicant stated that CCS system flow analysis is performed to demonstrate that the selected CCS pump head and flow characteristics ensure delivery of the required flow to all CCS users and also verify that the flow balancing orifices are sized with margin to be adjusted in the field. Also, an additional 7 percent margin in head is added to the CCS pump curve developed from the flow analysis, specifically to offset the effects of any degradation of pump performance occurring during the life of the plant. However, changes in CCS flow performance over time are expected to be minimal, since the CCS is a closed-loop, chemically-treated system with orifices used for flow balancing.

The applicant also stated that the CCS ITAAC requires a minimum flow rate of 10,164 Lpm (2685 gpm) to transfer heat from the RNS during shutdown. This flow rate, which assumes 10 percent degradation from the normal RNS HX flow requirement, is the minimum flow rate needed to remove decay heat from the RNS when it is aligned 4 hours after reactor shutdown (Mode 4). This flow rate must also be verified one day before entering Modes 5 and 6, as a SR included in CCS IPSAC, Table 16.3-2. Since a flow rate of 10,164 Lpm (2685 gpm) is sufficient to remove decay heat during Mode 4, it is also bounding for decay heat removal during Modes 5 and 6, when the RCS decay heat level has been further reduced. As a result, this SR will ensure that the CCS will be able to adequately perform its RTNSS function. This CCS minimum flow rate of 10,164 Lpm (2685 gpm) to the RNS HX is added to Tier 2 Table 9.2.2 1 in the DCD markup, as well as a similar 10 percent degraded value of 4543 Lpm (1200 gpm) to the SFS HX.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.2-SBPA-06 and finds that it adequately explains the CCS basis for flow required for its RTNSS functions with design adequate margins at 10 percent degraded values. The staff also reviewed the proposed changes to DCD Table 9.2.2-1 and finds that the table, as changed, adequately describes the basis for CCS flow rates. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant also proposes to increase the heat transfer capability of each CCS HX as specified in ITAAC Table 2.3.1-2. The current acceptance criterion specifies a UA value of 740×10^6 W/°C (12.1×10^6 Btu/hr-°F), and the applicant proposes to change this UA value to 856×10^6 W/°C (14.0×10^6 Btu/hr-°F). The value that is proposed as the ITAAC acceptance criterion is consistent with the revised value that is proposed in Tier 2, Table 9.2.2-1, and it is acceptable from this perspective. However, the applicant did not identify how the proposed CCS HX UA value was determined and how much margin is available to address operational considerations, on what basis this determination is appropriate and justified, and how the specified CCS heat transfer capability will be maintained over time. In RAI-SRP9.2.2-SBPA-07 the staff asked the applicant to address these heat transfer issues.

The applicant responded to RAI-SRP9.2.2-SBPA-07 in a letter dated June 26, 2008. The applicant provided additional information primarily related to CCS HX design and made reference to TR-111 for additional discussion. However, the response did not address the specific question that was asked.

During the June 25, 2009 audit, the staff explained that in order to resolve the RAI the applicant should provide more detail explaining how the proposed CCS HX coefficient of heat transfer (U) and required heat-transfer area (A) values were determined and how much margin is available to address operational considerations.

In a letter dated August 31, 2009, the applicant stated that the CCS HX UA was established to ensure that supply temperature did not exceed the RCP external HX cooling requirements, under maximum safety wet bulb temperatures. The design UA also bounds the UA value needed to meet the CCS temperature requirements for cooling down the RCS to cold shutdown conditions within 96 hours of reactor shutdown. Selecting a CCS HX UA, which meets temperature requirements during plant cooldown also ensures that the CCS HXs will be able to perform their DID and RTNSS functions of providing cooling to the RNS HXs during RCS cooldown and reduced reactor coolant inventory operation.

The staff reviewed the applicant's additional information in its response to RAI-SRP9.2.2-SBPA-07 and finds that it adequately explains how the CCS HX UA values were determined and how much margin is available to address operational considerations. The CCS HX specification requires the inclusion of additional heat transfer area above the design value to account for fouling and degradation over the HX's operating life. Additional frame length is also included so that additional plates can be added to the HX if additional performance is required. The HX margins are also discussed in Section 9.2.1 of this report and have been adequately addressed. Therefore, RAI-SRP9.2.2-SBPA-07 is resolved.

<u>CCS Pump Design Considerations</u>

Tier 2 Table 9.2.2-1 included additional proposed changes that had not been explained and justified. In particular, the bases for the proposed changes to the CCS pump design capacity and total developed head had not been addressed. Also, the bases for the proposed changes to the CCS HX design duty, design UA, and design flow rate (CCS side) had not been addressed. In RAI-SRP9.2.2-SBPA-08, the staff asked the applicant to address these heat transfer issues.

The applicant responded to RAI-SRP9.2.2-SBPA-08 in a letter dated June 26, 2008. The applicant indicated that the increased CCS pump design capacity is primarily due to increased cooling water flow requirements for the RCPs. The total developed head requirement for the CCS pumps was reduced substantially by increasing the diameter of several of the CCS main supply and return headers to minimize dynamic losses in the system that would otherwise result from the increase in CCS flow rate. The staff agrees that these particular changes are appropriate and justified for the reasons stated. However, the applicant's response did not adequately address and justify the proposed changes to the CCS HX design parameters.

During the June 25, 2009 audit, the applicant stated that the bases of the proposed CCS changes were discussed in TR-111. Further, the applicant clarified that other CCS changes (including lower CCS pump TDH, increasing piping sizes, and the reduction in flow velocities), were provided in TR-111.

The staff finds, based on the review of TR-111, that the UA of the CCS HX has been increased to meet all the CCS performance requirements with the increased heat loads for cooled components. The increase in CCS HX size is associated with an increased SWS flow rate as well as an increased CCS flow rate. On this basis the staff finds that RAI-SRP9.2.2-SBPA-08 is resolved.

CCS Cooling for RCPs, Instrumentation and Controls

In DCD Section 9.2.2.3.4, the applicant proposed to change the RCP component cooling water discharge isolation valves from MOVs to AOVs. The applicant changed the closing signal input to these valves to an AOV close signal generated by the plant control system. The close signal will use simultaneous CCS flow deviations in the RCP’s component cooling water supply and return lines. The applicant deleted reference to a tube rupture in the RCP motor cooling coil or thermal barrier as a source of reactor coolant leakage into the CCS and replaced this source of potential leakage with the RCP external HX. The staff finds that the proposed change does not eliminate the requirement that the RCP’s CCS outlet line be protected from overpressure by relief valves.

In DCD Section 9.2.2.4.5.2, the applicant further described the flow sensors in the CCS inlet and outlet lines associated with each RCP external HX and added that the cooling water outlet line isolation valves on each RCP are nonsafety-related. With the design change to a canned RCP, the applicant replaced the pump motor cooling coil and thermal barrier with an RCP external HX. The applicant also modified how the RCP component cooling water isolation signal is developed and further clarified the alarms an operator would receive.

In DCD Section 9.2.2.7, the applicant provided detail for the high-level and low-level alarms instrumentation on the CCS surge tank. There are two redundant level channels in the design to reduce the likelihood of reactor trip caused by a single downscale failure of a surge tank level channel. Such redundancy could preclude unnecessary tripping of CCS pumps which would subsequently cause loss of cooling flow to the RCPs and other cooled components. The CCS surge tank makeup water valve is automatically actuated by one of the two level channels, in order to provide makeup flow from the demineralized water transfer and storage system into the CCS. The applicant clarified that flow alarms in the MCR, produced by the two flow channels located on the CCS RCP cooling water inlet and outlet lines, will be used to alert the operator that a leak exists on the RCP external HX. The applicant identified that flow-measuring instrumentation on the RCP component cooling water inlet and outlet lines provides an isolation signal to close an AOV and isolates the leaking RCP external HX from the rest of the CCS.

The staff finds that changes to CCS valves, flow sensors, isolation signals and instrumentation provide an additional level of system reliability and do not result in negative or adverse system interactions. Based on its evaluation, the staff concludes that these valve and instrumentation changes are acceptable and do not change the NUREG-1793 Section 9.2.2 findings or conclusions.

9.2.2.2.2 Proposed Increase in the Maximum CCS Supply Temperature

Tier 2 of the AP1000 DCD, Section 9.2.2.1.2.1, "Normal Operation," proposes to increase the maximum allowed CCS supply temperature to plant components from 35 °C (95 °F) to 37.2 °C (99 °F) during normal plant operations, but the basis for this proposed change was not explained and justified. In RAI-SRP9.2.2-SBPA-09 the staff requested that the applicant justify this change.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant explained that the increased CCS supply temperature was due to the proposed increase in the maximum safety non-coincident wet bulb temperature from 27.2 °C (81 °F) to 29.7 °C (85.5 °F) using a cooling tower that is designed for an approach to wet bulb temperature of 13.3 °C (8 °F). The applicant referenced page 11 of TR-108 for additional explanation. The applicant also provided information regarding a further increase in the maximum safety non-coincident wet bulb temperature that was being made (but not yet submitted) to accommodate the Levy site parameters. This information was not included within the scope of this evaluation. While the information that was provided by the applicant explains to some extent how the maximum service water supply temperature is achieved, it did not explain how the maximum CCS supply temperature was determined and justified. The staff also noted that the use of non-conservative temperature assumptions for the plant shutdown and refueling heat transfer analyses was not explained and justified. Furthermore, this approach was not consistent with the information provided in Tier 2 Section 5.4.7.1.2.3, "In-Containment Refueling Water Storage Tank Cooling," which indicates that the maximum safety non-coincident wet bulb temperature is assumed for normal conditions and transients that start at normal conditions.

During the June 25, 2009 audit, the applicant clarified that the normal wet bulb is a realistic value for evaluating DID on investment protection as stated in its May 13, 2009, supplemental response to RAI-SRP9.2.1-SBPA-05. The maximum safety wet bulb temperature of 30.0 °C (86.1 °F) is applicable for full power operations as stated earlier in the May 7, 2009, and May 13, 2009, supplemental RAI responses. The higher ambient temperatures 30.0 °C vs. 26.7 °C (86.1 °F vs. 80.1 °F) will not impact safety or investment protection and would only result in an extended time to achieve cooldown. The staff agreed that the information was previously adequately presented; therefore, RAI-SRP9.2.2-SBPA-09 is resolved. Wet bulb temperature considerations are also discussed in Section 9.2.1 of this report and have been resolved.

In DCD Section 9.2.2.1.2.1 Revision 17, the applicant proposed a change in the maximum CCS supply temperature to plant components from 37.2 °C (99 °F) to 37.8 °C (100 °F). Additionally, the applicant proposed to raise the wet bulb temperature for service water cooling at normal operations (maximum normal temperature per Tier 2 Table 2-1 for normal shutdown). Although the assumption of a 0 percent exceedance was not changed, the ambient design wet bulb temperature would be raised from 29.7 °C (85.5 °F) to 30.0 °C (86.1 °F).

In APP-GW-GLE-036, the applicant described the impact of changing the current maximum wet bulb non-coincident temperature from 29.7 °C (85.5 °F) to 30.0 °C (86.1 °F) and the maximum wet bulb coincident temperature from 26.7 °C (80 °F) to 30.0 °C (86.1 °F) to encompass more sites in the eastern United States. In APP-GW-GLE-036, the applicant performed an evaluation of the effect on CCS to determine if sufficient margin exists to accommodate a 3.3 °C (6.1 °F) change and a 0.3 °C (0.6 °F) for both coincident and non-coincident wet bulb temperatures:

The applicant performed a design assessment and identified the following areas associated with CCS that are affected by the increased maximum wet bulb temperature:

- Safety system design basis – additional cases for containment analysis were included in the safety analysis to support the revised coincident and non-coincident wet bulb temperature.
- Normal, decay and SFP heat removal (cases relying on use of the 0 percent exceedance wet bulb temperature only)
- Component cooling and service water design

Safety System Design Basis

The applicant stated that no changes to the AP1000 design are needed to accommodate any safety issues because evaluations have demonstrated that the current AP1000 accident analyses will bound the revised coincident 46.1°C/30.0 °C (115 °F/86.1 °F) and non-coincident 30.0 °C (86.1 °F) wet bulb temperatures. The applicant stated that the maximum containment peak pressure performance of the passive containment cooling system at the higher wet bulb temperature is bounded by the current analysis for which a bounding sensitivity was documented in the Nuclear Safety Containment Analysis for AP1000.

Normal, Decay and Spent Fuel Pool Heat Removal

The applicant evaluated the impact of wet bulb temperature change on the performance of the normal RNS, SFP cooling, CCS and SWSs. The applicant stated that the performance evaluations considered normal operating modes, the ability to meet post shutdown cooldown times, full core offloads, loss of all ac power, and heat up of the IRWST.

The applicant stated that all design criteria were met including cooldown times and temperature limits with the exception of normal plant power operation with maximum heat loads, one CCS train in service, and at the maximum safety temperature limit of 30.0 °C (86.1 °F) wet bulb. The applicant identified that the exception exists for less than 30 hours per year and that with only one train of CCS, the CCS temperature would rise above 35.0 °C (95 °F) and then return to less than 35.0 °C (95 °F) by the time the 1 percent exceedance temperature was reached. The applicant identified the RCP motor cooling system as the most limiting component, which was designed to operate for at least 6 hours duration with a temperature up to 37.8 °C (100 °F). The applicant's evaluation was that with the maximum allowable cooling water temperature of 36.1 °C (97 °F) for the RCPs (the most limiting component), this change was acceptable.

Component Cooling (CCS) and Service Water System (SWS) design

The applicant stated that CCS will accommodate the heat loads from operations without impacting performance or sizing of the CCS. The SWS cooling tower is not expected to require changes to accommodate the higher wet bulb temperatures since sizing is based on plant cooldown at 4 hours after reactor shutdown, and is based on the unchanged 1 percent exceedance value of 26.7 °C (80.1 °F). The applicant explained that the SWS cooling water supply temperature for the maximum safety case will be 33.1 °C (91.6 °F), which will result in a maximum CCS supply temperature of 36.1 °C (97.0 °F).

In APP-GW-GLE-036, the applicant states that the most limiting CCS component is the RCP motor cooling system and that temperatures of up to 37.8 °C (100 °F) for a duration of 6 hours are acceptable. In DCD Revision 17 Table 5.4-1, the RCP maximum continuous component cooling water inlet temperature is given as 35 °C (95 °F) with a 6 hour elevated temperature of up to 43.3 °C (110 °F). Additionally the applicant identified in the DCD that an input to a reactor trip is RCP “Hi Bearing Temperature” but there was no mention of high motor temperature. As a result of the design change to canned RCPs, the staff asked the applicant, in RAI-SRP9.2.2-SBPA-14, to verify that the RCP motor cooling system is still the most limiting CCS supply temperature: if the RCP motor cooling system is no longer the most limiting CCS cooled component, identify and provide the evaluation of the impacts of the revised wet bulb temperature limit on the plant for the new limiting component. The staff also asked the applicant, in RAI-SRP9.2.2-SBPA-14, to clarify a statement in APP-GW-GLE-036.

The applicant responded to RAI-SRP9.2.2-SBPA-14 on August 31, 2009 and stated that the RCP motor cooling system is still the most limiting component served by the CCS with respect to maximum temperature of the supplied cooling water. The limiting CCS supply temperature for RCP cooling is 37.7 °C (100 °F). The RCP can operate at full speed with CCS supply temperature at this level for up to 6 hours continuously. Since the CCS and SWS are both designed with significant thermal margin, the actual CCS supply temperature to the RCPs and to other CCS components with the plant at power will always be lower than the limiting value of 37.7 °C (100 °F), which assumes maximum operating heat load on the CCS, 4 °C (8 °F) cooling tower approach to wet bulb, and local ambient wet bulb temperature at the maximum safety (0 percent exeedence) level. During plant cooldown with RCPs operating, the CCS temperature may approach 37.7 °C (100 °F) for a few hours at the highest plant cooldown rate of 27.8 °C/h (50 °F/h), but the RCPs are operating at reduced speed in this mode and their cooling requirements are therefore less stringent than for full power, full speed operation.

The staff review determined that this response is acceptable since the design conditions of 37.7 °C (100 °F) have been established in DCD Section 9.2.2.1.2.1, “Normal Operations”. During the cooldown period, the component cooling water inlet temperature to the various components does not exceed 43.33 °C (110 °F) as described in DCD Section 9.2.2.4.3, “Plant Shutdown” which is consistent with Table 5.4-1. For this reason, the staff determined RAI-SRP9.2.2-SBPA-14 is resolved.

9.2.2.2.3 Revised Site Interface Temperature Limits

As discussed in TR-108, the applicant proposes to change the site interface temperature limits to encompass a broader range of potential sites for AP1000 plants. In particular, Tier 1 of the DCD, Table 5.0-1, “Site Parameters,” would be revised to specify a maximum (noncoincident) wet bulb temperature of 29.7 °C (85.5 °F) instead of 27.2 °C (81 °F). Tier 2 of the DCD, Table 2-1, “Site Parameters,” would be revised to reflect this higher (noncoincident) wet bulb temperature as the Maximum Safety (or 0 percent exceedance) value, and the Maximum Normal (or 1 percent exceedance) wet bulb temperature would be revised from 25 °C (77 °F) to 26.7 °C (80.1 °F) for the coincident value and from 26.7 °C (80 °F) to 26.7 °C (80.1 °F) for the noncoincident value.

The proposed changes to the site interface temperature limits are reflected in Tier 2 of the AP1000 DCD, Section 9.2.2, in place of the values that were previously listed. Although the values correspond to how they were used previously (i.e., the replaced values are “like-for-like”), the Tier 2 description does not explain why the maximum safety (noncoincident) wet bulb temperature is specified for normal operation and the maximum normal wet bulb temperature is specified for other cases. It is not clear why the maximum safety limit does not apply for the CCS DID and RTNSS functions. The staff asked the applicant, in RAI-SRP9.2.2-SBPA-10, to explain and justify this approach and to revise the Tier 2 information to clearly describe the plant design basis in this regard.

The applicant responded to RAI-SRP9.2.2-SPBA-10 in a letter dated June 26, 2008. The applicant explained that the maximum safety non-coincident wet bulb temperature does not apply to RTNSS and Investment Protection functions because they are not functions required to guarantee the safety of the plant. However, contrary to this logic, the applicant also explained that the maximum safety non-coincident wet bulb temperature is used in determining CCS and SWS performance for power operation since the peak ambient wet bulb temperature has a relatively high likelihood of occurrence during the operating portion of a refueling cycle. The applicant failed to recognize that elevated temperature conditions tend to increase the likelihood of plant trips and transients due to grid instability and assurance needs to be provided that DID and RTNSS SSCs are capable of performing their functions whenever the maximum normal wet bulb temperature is exceeded. The applicant's response did not address the staff's concerns in this regard.

During the June 25, 2009 audit, the applicant clarified that, as stated in its response to RAI-SRP9.2.1-SBPA-05, with respect to the 26.7 °C (80.1 °F) wet bulb for RTNSS, normal cooldown can be accomplished sooner than it can be accomplished with the 30.0 °C (86.1 °F) wet bulb. Further, the applicant clarified that all the temperature limits have margins. The applicant explained that passive safety systems are not needed. Further, the applicant clarified that the normal wet bulb temperature is a realistic value for evaluating DID on investment protection. The maximum safety wet bulb temperature of 30.0 °C (86.1 °F) is applicable for full power operations, as previously presented in the RAI-SRP9.2.2-SBPA-09 response dated May 7, 2009.

The staff agreed that the information was previously adequately presented; therefore, RAI-SRP9.2.2-SBPA-10 is resolved. The higher ambient temperatures 30.0 °C vs. 26.7 °C (86.1 °F vs. 80.1 °F) will not impact safety or investment protection and would only result in an extended time to achieve cooldown.

9.2.2.2.4 Use of Nonmetallic Pipe

The applicant proposed a change to Tier 2 of the DCD, Section 9.2.2.3.5, "Piping Requirements," to allow COL applicants the option of using black polyethylene piping (High Density Polyethylene or HDPE) for CCS applications in accordance with the ASME B31.1 Power Piping Code if deemed appropriate by evaluation. In particular, HDPE could be used in areas of low pressure and low temperature, up to 1,034 kPa (150 psi) and 60 °C (140 °F). The basis for the use of nonmetallic pipe for this application is described in TR-103.

Although the CCS is subject to RTNSS, it is not relied upon for post-72 hour cooling following an accident and the design provisions that pertain to seismic, flooding, and hurricane conditions do not apply. Therefore, from this perspective, the proposed use of HDPE is acceptable. However, since the CCS function is risk important during shutdown conditions when the reactor is open, the impact of using HDPE on CCS reliability and availability assumptions needs to be considered and addressed. Also, the review criteria specified by NUREG-0800 Section 3.6.1 relative to pipe failure evaluations is based on the use of metal pipe. Unless otherwise justified by the applicant, the potential consequences of pipe failure (including flooding) should be evaluated assuming the complete failure of all HDPE piping during seismic events coincident with metallic pipe failures that are postulated and other considerations that are specified by the SRP. Finally, the specific criteria for allowing the use of HDPE should be specified in the DCD to ensure clarity of the plant licensing basis. The applicant was asked, in RAI-SRP9.2.2-SBPA-11, to revise the DCD (Tier 1 and Tier 2 as appropriate) to address these considerations. The applicant responded to the staff's request in a letter dated June 26, 2008, and referred to its earlier response to RAI-TR103-EMB2-02 dated February 22, 2008. The applicant also indicated that HDPE is not used in the AP1000 CCS design and that there are no current plans to use HDPE in this system. However, because use of HDPE is proposed as an option for COL applicants, its use needed to be fully evaluated and justified by the applicant. The response that was provided by the applicant did not provide the information that was requested in RAI-SRP9.2.2-SBPA-11. As a separate matter, the staff also requested that the applicant describe how the requirements specified by 10 CFR 20.1406 are satisfied with respect to CCS considerations, including provisions that have been established for buried or inaccessible pipe.

During the June 25, 2009 audit, the staff explained that in order to resolve the RAI the applicant should provide more detail explaining the use of nonmetallic piping in the CCS.

In a letter dated August 31, 2009, the applicant stated that the provision for the use of nonmetallic piping is removed from DCD Tier 2 Section 9.2.2. The operating pressure and temperature for the CCS exceeds the limits for HDPE piping imposed by ANSI/ASME B31.1 and applicable code cases and is, therefore, prohibited for use in this application. Additionally, the applicant proposed to revise DCD Tier 2 Section 9.2.2.3.5 "Piping Requirements" to delete references to nonmetallic piping. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Related to 10 CFR 20.1406, in the applicant's submittal of APP-GW-GLN-098 dated April 10, 2007 and the applicant response to RAI-SRP-12.1-CHPB-01 dated September 9, 2008 all radioactive piping is located inside the auxiliary building, which minimizes the potential for leakage to the groundwater from piping and fittings. In addition, no piping containing radioactive fluid is directly buried in the ground. In addition, the use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA as described in DCD Section 12.3.1.1.2, "Common Facility and Layout Designs of ALARA". To the extent possible, pipes are routed in accessible areas such as dedicated pipe routing tunnels or pipe trenches; this provides good conditions for decommissioning. Based on the staff's review, the staff determined 10 CFR 20.1406 has been adequately addressed since the CCS is not buried and radiation monitors, which monitor RCS leakage into CCS, alarm in the MCR. 10 CFR 20.1406 design considerations are further discussed in Chapter 12 of this report.

In DCD Section 9.2.2.3.5 Revision 17, the applicant corrected the code reference applicable to nonmetallic piping. The applicant changed the specification for nonmetallic piping from "used in accordance with ASME B31.1" to "constructed to the requirements of ANSI B31.1 Appendix III" and limited the use of nonmetallic piping to outside containment for the CCS system.

Based on its evaluation, the staff finds that this change limits the use of nonmetallic piping to areas outside containment which are outside the risk-important portions of the CCS function during shutdown conditions. Based on its evaluation, the staff concludes that these changes are acceptable and do not change the NUREG-1793 Section 9.2.2 findings or conclusions related to CCS piping requirements.

9.2.2.2.5 Investment Protection Short-Term Availability Controls

The applicant proposed to change the minimum required CCS flow rate that is specified for the normal shutdown cooling HX in DCD Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," SR 2.3.1. This SR is revised to specify that each CCS pump needs to provide at least 10,164 Lpm (2685 gpm) through a normal shutdown cooling HX, which is consistent with the flow rate specified in ITAAC Table 2.3.1-2 for Design Commitment 3 (it is noted that a change is not being proposed for the ITAAC value that was originally established). However, SR 2.3.1 previously specified a minimum flow rate of 10,675 Lpm (2820 gpm), and it was not clear why the ITAAC value that was established was not the same as the value that was originally specified by SR 2.3.1 and why the ITAAC value was correct.

In RAI-SRP9.2.2-SBPA-12, the applicant was asked to explain this apparent inconsistency and to adequately justify the proposed change to reduce the minimum flow rate specified in IPSAC SR 2.3.1 in order for the staff to determine if the proposed change was acceptable.

The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant indicated that 10,675 Lpm (2820 gpm) is the normal CCS flow rate to each of the shutdown cooling HXs and a flow rate of this value or higher is expected to be achieved with the CCS configured as required to perform the normal shutdown cooling function related to CCS flow (i.e., each CCS pump supplying one shutdown cooling HX, one spent fuel cooling HX, and CCS auxiliary loads). The value of 10,164 Lpm (2685 gpm) in DCD Tier 1, Table 2.3.1.2 represents the minimum required CCS flow rate to accomplish the shutdown cooling and is therefore the flow that must be demonstrated in the ITAAC.

The staff reviewed the changes in flow rate values and determined they were adequately explained since the value of 10,164 Lpm (2685 gpm) represents the 'minimum' required flow rate to accomplished the shutdown cooling and the value of 10,675 Lpm (2820 gpm) represents the 'normal' CCS flow rate to perform a 'normal' shutdown cooling function. Since the Tier 1 ITAAC and the SRs have consistent values, the staff determined that RAI-SRP9.2.2-SBPA-12 is resolved.

9.2.2.2.6 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Considerations

The applicant proposed to change the CCS flow rate and heat removal capability that are specified in ITAAC Table 2.3.1-2 to reflect proposed changes that are discussed above in Section 9.2.2.1.1. The changes that are proposed for the SWS ITAAC are consistent with the specific changes that are proposed in this section.

9.2.2.2.7 Initial Test Program Considerations

The initial test program for the CCS is discussed in Tier 2 of the DCD, Section 14.2.9.2.5, "Component Cooling Water System Testing." The stated purpose of the CCS test program is to verify that the as-installed CCS performs the DID functions described in DCD Tier 2 Section 9.2.2 of providing cooling water to DID components and transfer heat to the SWS; as well as providing cooling water to other non-safety-related components for heat removal.

9.2.2.3 Conclusion

The staff evaluated proposed changes to Revision 15 of the AP1000 DCD that pertain to the CCS. The proposed changes are documented in Revision 17 of the AP1000 DCD Tier 2, Section 9.2.2, and are reflected in the Tier 1 ITAAC specified in Table 2.3.1-2 and the Tier 2 IPSAC specified in Table 16.2, Section 2.3. The staff's evaluation, using the guidance provided by NUREG-0800 Section 9.2.2, confirmed that: a) the proposed changes will not adversely affect safety-related SSCs; b) the SWS is capable of performing its DID and RTNSS functions; and c) ITAAC, IPSAC, and initial test program considerations are adequate and appropriate. The proposed changes conform to the existing AP1000 licensing basis as documented in Revision 15 of the approved DCD. The changes contribute to the increased standardization of this aspect of the design. Therefore, these changes meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii). The staff finds that the proposed changes are acceptable.

9.2.5 Potable Water System

9.2.5.1 Summary of Technical Information

Section 9.2.5, "Potable Water System," (PWS) of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In AP1000 DCD, Revision 17, the applicant has proposed to delete the site-specific PWS design, including the supply source, from the scope of the certified design and to add a proposed COL information item to address this information.

The basis for this change is documented in TR-124, "Removal of PWS Source and WWS Retention Basins from the Westinghouse AP1000 Scope of Certification," APP-GW-GLN-124, Revision 0 of June 2007. The applicant has identified this change in AP1000 DCD Tier 2, Revision 17, Section 9.2.5.

9.2.5.2 Evaluation

The staff reviewed all changes to the PWS in AP1000 DCD Revision 17 in accordance with the guidance in NUREG-0800 Section 9.2.4, "Potable and Sanitary Water Systems." The staff did not re-review descriptions and evaluations of the PWS in AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. The regulatory basis for Section 9.2.5 of the AP1000 DCD is documented in NUREG-1793. The specific criterion that applies to the proposed DCD change is 10 CFR 52.63(a)(1)(vii) which concerns contribution to increased standardization of the certification information.

In TR-124, "Removal of PWS Source and WWS Retention Basins from the applicant AP1000 Scope of Certification", Revision 0, the applicant proposes to remove the PWS source from the certified design including the potable water storage tank, potable water pumps, potable water jockey pump, and their associated piping. The applicant proposes to transfer responsibility for addressing the supply source and components of the PWS outside of the power block to the COL applicant through the addition of COL Information Item 9.2.11.1. Proposed COL Information Item 9.2.11.1 in DCD Revision 17 states:

> The Combined License applicant will address the components of the potable water system outside of the power block, including supply source required to meet design pressure and capacity requirements, specific chemical selected for use as a biocide, and any storage requirements deemed necessary. A biocide such as sodium hypochlorite is recommended. Toxic gases such as chlorine are not recommended. The impact of toxic gases on the main control room compatibility is addressed in Section 6.4.

The staff identified a wording error in the last sentence of this proposed COL Information Item where "control room *habitability"* should be used in place of the proposed "control room *compatibility."* The applicant agreed during the March 18, 2009 Public Meeting that the correct wording is "control room *habitability."* The staff finds that the proposed COL information item adequately addresses the necessary information that was removed from the DCD by this change. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In DCD Revision 17, the applicant states that no interconnections exist between the PWS and any potentially radioactive system or any system using water for purposes other than domestic water service. The changes to the PWS include the removal of portions of the PWS, and providing site specific information in COL Information Item 9.2.11.1. Because the changes are all outside the power block and do not involve possible contamination by radioactive water, the staff finds that the conclusions of NUREG-1793 Section 9.2.5 remain valid. Specifically, the staff finds that the PWS continues to satisfy GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to design provisions for controlling the release of water containing radioactive material and preventing contamination of the potable water.

9.2.5.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the applicant's application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 PWS as documented in DCD, and in TR-124. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 PWS to meet the applicable NUREG-0800 Section 9.2.4 acceptance criteria. The staff also finds that the design changes have been properly incorporated into the appropriate sections of AP1000 DCD, Revision 17. The AP1000 PWS design continues to meet all applicable acceptance criteria, and the proposed change meets the criterion of 10 CFR 52.63(a)(1)(vii), on the basis that it contributes to increased standardization of the certification information. The staff finds that all of the changes related to the system design of the AP1000 PWS are acceptable.

9.2.7 Central Chilled Water System

9.2.7.1 Summary of Technical Information

Revision 17 of the AP1000 DCD includes proposed changes to the VWS in order to specify increased wet and dry bulb temperatures for heat load considerations and to provide additional design flexibility for COL applicants. The basis for these proposed changes are discussed in TR-108, TR-103 and TR-107, "AP1000 Technical Support Center," APP-GW-GLR-107, Revision 1, of June 2007.

9.2.7.2 Evaluation

The containment isolation interface of the VWS is safety-related. The balance of the VWS is nonsafety-related and the regulatory basis for evaluating the safety and non-safety systems is documented in Section 9.2.7, "Central Chilled Water System," of NUREG-1793. Although the VWS is nonsafety-related, the low-capacity subsystem is considered to be important to safety because it provides chilled water for cooling safety-related and DID equipment rooms. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs. In addition, the staff evaluation focused on those items that satisfy the criteria for RTNSS, the capability of the VWS to perform its RTNSS and DID cooling functions, and the adequacy of ITAAC, test program specifications, and RTNSS availability controls that have been established for the VWS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was judged based upon conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.2 (as applicable), and SECY-94-084.

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

During its evaluation, the staff noted that the description that is provided in Revision 17 of the DCD, Section 9.2.7, does not describe the DID and investment protection functions of the VWS. However, the ITAAC specified in Tier 1 of the DCD, Section 2.7.2, "Central Chilled Water System," the initial test program described in Tier 2 of the DCD, Section 14.2.9.2.9, "Central Chilled Water System Testing," and Table 17.4-1, "Risk-Significant SSCs Within the Scope of D-RAP," indicate that the VWS is important for both DID and investment protection considerations. The staff found that this information was not reflected in the description that is provided for the VWS in Tier 2 of the DCD, Section 9.2.7, and that no investment protection short-term availability controls were established for this system. In RAI-SRP9.2.2-SBPA-02, the applicant was asked to provide clarification as necessary in the AP1000 DCD to better explain the DID and investment protection functions of the VWS, as well as to explain why IPSAC was not necessary recognizing that VWS is needed to support other DID non-safety systems. The applicant responded to the staff's request in a letter dated June 26, 2008. The applicant provided additional information concerning the DID and investment protection functions provided by the VWS, and similarly for IPSAC considerations. The staff determined that the additional information was incomplete and inadequate, and Tier 2 of the DCD was not revised to fully explain the VWS design basis relative to DID and investment protection considerations.

Following the June 25, 2009 audit, the applicant responded on September 4, 2009, with an RAI response revision that clarified that the VWS itself is not captured in the RTNSS program. The applicant clarified the functions of the low capacity subsystem that are to maintain the MCR, 1E electrical room and normal RNS pump rooms room temperatures and the associated concrete heat sink temperatures. The applicant identified components of the VWS that are determined to be risk-significant and are included in the scope of D-RAP. These components are identified as:

- Two pumps associated with the low capacity subsystem (as shown on DCD Tier 2 Figure 9.2.7-1 (Sheet 1), "Central Chilled Water System Piping and Instrumentation Diagram".

- Two air-cooled chillers associated with the low capacity subsystem (as shown on DCD Tier 2 Figure 9.2.7-1 (Sheet 1).

In addition the applicant provided a DCD markup adding the above items to Tier 2 DCD Section 9.2.7.2.2, "Component Description," including references to DCD Table 17.4-1, "Risk-Significant SSCs within the Scope of D-RAP." Also included in the DCD markup was a simplified drawing describing the high capacity subsystem (which was omitted in Revision 17 of the DCD) and clarification to DCD Table 9.2.7-1, "Component Data-Central Chilled Water

System," related to clarification of components between the high capacity subsystem and the low capacity subsystem.

The staff reviewed the material from the audit, RAI revision response, and DCD markup and concluded that the clarification to the D-RAP components was adequately addressed and the staff verified that there were no technical changes made to the DCD. In addition, the staff finds that the RAI response corrects the inconsistency between Section 9.2.7 of the DCD and Table 17.4-1. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.2.7.2.1 Impact of Proposed Changes on Safety-Related SSCs and Functional Capability

Increase in Site Interface Temperature Limits

As discussed in TR-108, the applicant proposes to change the site interface temperature limits to encompass a broader range of potential sites for AP1000 plants. The proposed changes for the VWS are reflected in Tier 2 of the DCD, Section 9.2.7.2.4, "System Operation." These changes relate to the design specifications for the VWS and do not result in new VWS failure modes or interactions that can adversely affect the capability of safety-related SSCs to mitigate postulated accident conditions. Also, TR-108 states that the limiting temperature specifications are used for properly sizing the air-cooled chiller, thereby assuring adequate DID cooling capability for the low-capacity subsystem. Therefore, as discussed in Section 9.2.2.2.3 of this report, the proposed changes to the site temperature interface limits are acceptable.

Removal of Smart Valves

The AP1000 design specifies the use of "smart valves" (i.e., valves that contain instrumentation such as temperature, flow and pressure that is used for control or indication) for some system applications. In the case of the VWS, smart valves (V272A/B and V261A/B/C/D) are specified as the modulating control valves. As discussed in TR-103, the applicant proposes to remove the requirement for using smart valves for this function so that standard valves can be used by COL applicants. The proposed change replaces the instrumentation that is included in the smart valve design with standard inline instrumentation as illustrated in Figure 9.2.7-1 of the DCD Tier 2. The staff determined the proposed change does not eliminate or alter the functional capabilities of any VWS valves or instruments, and will not degrade the capability or reliability of the VWS to perform its function. The staff expects this change to improve the capability to service and maintain the affected instrumentation which would tend to improve VWS availability and reliability consistent with the Commission's policy on RTNSS. Therefore, the staff considers the proposed changes to use standard inline instrumentation to be acceptable.

Design Temperature inside Containment

Tier 2 of the AP1000 DCD Revision 17, Section 9.2.7.2.2, modified the piping inside containment for the nonsafety high capacity chilled water system from a design temperature of 160 °C to 93.3 °C (320 °F to 200 °F) to accommodate both cooling and heating service. The staff finds that lowering the design temperature of the CWS piping inside containment is a reduction in conservatism. For this reason, the staff discussed this issue at the June 25, 2009, audit and issued RAI-SRP9.2.2-SBPA-13.

The applicant responded to this RAI on September 4, 2009 and stated the piping has a limit of 160 °C (320 °F) at 1,379 kPa (200 psig) and the applicant will restore the VWS piping design temperature back to 160 °C (320 °F). In addition, a DCD mark up was provided for Section 9.2.7.2.2.

The staff reviewed this response and finds that the proposed DCD change is acceptable since it returned to the design temperature based on its previous value. The higher temperature is based on the high capacity chilled water system being aligned to the hot water system for heating of containment. Since a DCD markup was provided as part of this RAI response, this item is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.2.7.2.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Considerations

Tier 1 of the AP1000 DCD, Section 2.7.2, “Central Chilled Water System,” specifies ITAAC for the VWS. The proposed changes referred to above include design considerations and editorial changes that do not alter or otherwise affect the design specifications that must be verified to assure the DID capability of the low capacity subsystem. Therefore, the staff finds that the ITAAC for VWS are not affected by the proposed changes and will continue to be acceptable.

9.2.7.2.3 Initial Test Program Considerations

The initial test program for the VWS is discussed in Tier 2 of the DCD, Section 14.2.9.2.9, “Central Chilled Water System Testing.” The purpose of VWS testing is primarily to verify that the as-installed low capacity subsystem adequately performs its DID cooling function, as well as confirming the proper function of the high capacity subsystem. The proposed changes referred to above do not alter the fundamental VWS performance considerations that apply and, therefore, the staff finds that the initial test program for VWS will continue to be acceptable.

9.2.7.3 Conclusion

The staff evaluated proposed changes to the VWS as discussed above and reflected in the AP1000 DCD, Tier 2, Section 9.2.9. The proposed changes involve a slight increase in the site temperature interface limits, elimination of smart valves from the VWS design, and design temperature changes inside containment. Based on the results of this evaluation, the staff has determined that the proposed changes will not adversely affect safety-related SSCs, the capability of the VWS to perform its DID cooling functions, or ITAAC and initial test program considerations. Consequently, the staff finds that the proposed changes are consistent with the AP1000 licensing basis and the applicable NRC review guidance specified by NUREG-0800 Section 9.2.2. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii), in that they contribute to increased standardization of the certification information; therefore, the proposed changes are acceptable.

9.2.8 Turbine Building Closed Cooling Water System

9.2.8.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.2.8, “Turbine Building Closed Cooling System,” of the AP1000 DCD, Revision 15. In the AP1000, Revisions 16 and 17 the applicant proposed changes to Section 9.2.8.

The applicant proposed the following technical changes to Revision 15 of the AP1000 DCD, which are supported by information in the TRs:

1. Remove descriptive information that pertains to the alternative steam and power conversion design that no longer applies, eliminate nominal heat load values from Table 9.2.8-1, "Turbine Building Closed-Cooling Water System Equipment Load List," and revise the title of the table accordingly, and increase the maximum temperature specifications for the turbine building closed cooling water system (TCS) and its heat sink. Change Table 9.2.8-1 and delete, from the equipment load list, the condensate pump motor air cooler, condensate pump bearing oil cooler, feedwater pump motor air cooler, and condenser vacuum pump.

2. Allow the use of non-metallic pipe in the system design, designate that the heat sink for the TCS is conceptual design information (CDI) instead of the circulating water system, to specify that backwashable strainers are provided upstream of the TCS HXs, and to include information related to TCS HX and upstream strainer operation. Clarify that nonmetallic piping may be used in the TCS and deleted the reference to ASME B31.1, "Power Piping".

The bases for these proposed changes are discussed in TR-86, "Alternate Steam and Power Conversion Design," APP-GW-GLN-018, Revision 1, dated June 2007, and TR-103.

9.2.8.2 Evaluation

The TCS is nonsafety-related and the regulatory basis for evaluating this system is documented in Section 9.2.8 of NUREG-1793. The staff's evaluation of the changes that are proposed focuses on confirming that the changes will not adversely affect SSCs or those that satisfy the criteria for RTNSS. The proposed changes were evaluated using the guidance provided by NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was judged based upon conformance with the existing AP1000 licensing basis and the guidance specified by NUREG-0800 Section 9.2.2 (as applicable).

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

<u>Impact of Proposed Changes on Safety-Related SSCs</u>

As discussed in Section 9.2.8 of NUREG-1793, TCS piping and components are located entirely within the turbine building. No safety-related equipment is located in the turbine building and, therefore, failure of the TCS cannot lead to the failure of any safety-related SSCs. On this basis, the staff found that the proposed change conforms to Regulatory Position C2 of RG 1.29, "Seismic Design Classification," thereby satisfying GDC 2 requirements and the applicable guidance of NUREG-0800 Section 9.2.2 with respect to impact on safety-related SSCs. The proposed changes referred to above and described in TR-86 and TR-103 relative to the TCS do not alter the location of the TCS relative to safety-related SSCs and, consequently, the basis for NRC approval in NUREG-1793 Section 9.2.8 remains valid in this respect.

The staff also noted that TR-103 (Page 23, Item 4) indicates that the cooling medium for the TCS HXs is changed from circulating water to a generic "cooling water" that can be provided by either circulating water and/or raw water makeup to the cooling tower basin. Section 9.2.8.1.2, "Power Generation Design Basis," describes the heat sink for the TCS as circulating water.

However, DCD Tier 2, Revision 17, Section 10.4.5.1.2, “Power Generation Design Basis,” for the circulating water system indicates that: “The CWS and/or makeup water from the raw water system supplies cooling water to the turbine building closed cooling water system (TCS) heat exchangers…” Consequently, the information provided in Sections 9.2.8 and 10.4.5 is inconsistent. It is not clear if the intent is to establish a CDI item for COL applicants to address or to provide the option of using the circulating water and/or raw water makeup to the cooling tower basin instead of establishing a CDI item. In RAI-SRP9.2.2-SBPA-01 the applicant was asked to provide additional information to explain the intention of the proposed change, and to revise Tier 2 of the DCD, Sections 9.2.8 and 10.4.5 as necessary to eliminate the inconsistency. The applicant indicated that the intent of the proposed change was not to establish a CDI item for COL applicants but rather to provide an option for COL applicants to utilize circulating water system (CWS) cooling tower makeup water flow or circulating water flow as the cooling water source, at the applicant’s discretion, for the TCS HXs. The applicant proposed an additional change to Tier 2 Section 10.4.5.2.2 of the DCD in order to eliminate this inconsistency.

The staff reviewed the proposed DCD markups and determined that the changes eliminated the inconsistency between DCD Tier 2 Sections 9.2.8 and 10.4.5. The staff verified that the proposed DCD markups in the RAI response were added to the DCD. Therefore, RAI-SRP9.2.2-SBPA-01 is resolved.

Impact of Removal of TCS Equipment Heat Loads from DCD Table 9.2.8-1

The applicant deleted equipment heat loads that were originally listed because the cooling water for the deleted equipment is a site-specific design rather than part of the AP1000 standard design.

Based on its evaluation, the staff finds this change acceptable because it does not affect the conclusions in NUREG-1793 Section 9.2.8.

Impact of TCS piping material clarification

The applicant deleted reference to ASME B31.1 and has added nonmetallic as a possible piping material to be utilized in the TCS.

The staff reviewed this clarification and determined that this change is consistent with Section 3.2.2.7, “Other Equipment Classes.” TCS is classified as ‘Class E’, and specific piping codes are not typically described in Section 3.2.2.7 for ‘Class E’ systems. In addition, the clarification to use nonmetallic piping in the TCS gives flexibility in the use of non-corrosive material as needed. In a corrosive water environment, nonmetallic piping material outperforms metallic materials whereas metallic piping material may require replacing over time. Based on the staff’s evaluation, the TCS piping material classification is acceptable.

9.2.8.3 Conclusion

The staff evaluated proposed changes to the TCS that are discussed above and are reflected in the AP1000 DCD, Tier 2, Revision 17, Section 9.2.8. Based on the results of this evaluation, the staff has determined that the proposed changes will not adversely affect safety-related SSCs, that they meet the criterion of 10 CFR 52.63(a)(1)(vii), in that they contribute to increased standardization of the certification information and are, therefore, acceptable.

9.2.9 Waste Water System

9.2.9.1 Summary of Technical Information

Section 9.2.9, "Waste Water System" of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant proposed to make several changes to Subsections 9.2.9.2.1, "General Description," 9.2.9.2.2, "Component Description," 9.2.9.5, "Instrumentation Applications," and 9.2.11.2, "Waste Water Retention Basin" of the certified design. The applicant also made changes to DCD Tier 1 Section 2.3.29, "Radioactive Waste Drain System," as a result of changes to the nonradioactive waste water system (WWS). All of these changes are related to the removal of the PWS from the scope of the DC, as described in TR-124.

9.2.9.2 Evaluation

The staff reviewed the proposed changes to Tier 2 Subsections 9.2.9.2.1, 9.2.9.2.2, 9.2.9.5, and 9.2.11.2, and Tier 1 Section 2.3.29, of the AP1000 DCD Revision 17, in accordance with the guidance of NUREG-0800 Section 9.3.3, "Equipment and Floor Drainage System." The regulatory basis for Section 9.2.9 of the AP1000 DCD is documented in Section 9.2.9 of NUREG-1793.

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

The staff finds that the proposed changes in Section 9.2.9.2.1 are limited to the removal of references to the waste water retention basin pump and transfer pumps since these are no longer part of the DC. In addition, the applicant removed the description of the condenser waterbox drains, which were previously routed to the waste water retention basin. The applicant stated that the design and routing of the condenser waterbox drains will be incorporated into the site-specific CWS design (discussed in Section 10.4.5 of NUREG-1793, as supplemented by this report). Because the design includes site-specific criteria, the staff finds acceptable that the applicant defers the design details of the condenser waterbox drains to the COL applicant. However, the staff determined that this information should be included in the COL information item described in DCD Tier 2 Section 10.4.12. The staff requested that the applicant include this information in the DCD in RAI-SRP9.3.3–SBPA-01.

In DCD Section 9.2.9.5, the applicant removed references to the level instrumentation and pump controls located in the waste water retention basin. The staff finds this change acceptable because it is not related to the guidance and acceptance criteria in NUREG-0800 Section 9.3.3.

In a letter dated June 20, 2008, the applicant responded to RAI-SRP9.3.3-SBPA-01 by proposing a markup of Section 10.4.12.1 of the DCD to state that the COL applicant will identify the action of routing the condenser waterbox drains with the site-specific CWS. The staff finds this markup acceptable because the applicant made the DCD clear with regard to what is within the scope of the DC. This DCD change was incorporated into Revision 17; RAI-SRP9.3.3-SBPA-01 is, therefore, resolved.

In DCD Section 9.2.9.2.2, the applicant removed the component description of the waste water retention basin and its associated basin transfer pumps. The applicant replaced this information with a reference to the COL information item described in Section 9.2.11, which requires the

COL applicant to provide the site-specific information for these components. Because the waste water retention basin and its associated basin transfer pumps and piping are site-specific components, the staff finds it acceptable to defer this to the COL applicant.

In DCD Section 9.2.9.5, the applicant also relocated the radiation monitor from the waste water retention basin to the turbine building sump. The staff reviewed this change to ensure that all effluents in the WWS that discharge to the turbine building sump will be monitored prior to disposition, as required by GDC 60. However, based on the information provided, the staff was unable to verify that all nonradioactive effluents will be monitored prior to disposition. For example, in DCD Revision 15, the condenser waterbox drains were routed directly to the WWS retention basin. The staff requested that the applicant address this in RAI-SRP9.3.3–SBPA-02.

By letter dated June 20, 2008, the applicant responded to RAI-SRP9.3.3-SBPA-02. In its response, the applicant identified all the sources of waste water that will drain downstream of the turbine building sump, which include diesel fuel area sump (upstream of the oil separator), SWS/CWS backwash, and other site specific effluent (e.g., CWS waterbox drain). The service water flow is provided with a radiation monitor. All systems interfacing with the CWS that have plausible potential for radioactive contamination are provided with radiation monitoring. The diesel fuel area sump effluent does not interact with any potentially radioactive sources during operation, nor are there any recognized radioactive sources located in the vicinity of the portion of WWS. Effluents that are site specific are under the responsibility of the COL applicant to ensure proper radiation monitoring is designed into the system, as noted in COL Information Item 11.5-1 (Section 11.5.7 of the AP1000 DCD). Based on the above, the staff concludes that all potentially radioactive effluents from the standard plant are properly monitored for radiation prior to disposition off site, as required by GDC 60.

Based on the evaluation of the DCD information and the above response to RAI-SRP9.3.3-SBPA-02, the staff finds this change acceptable because it does not affect the NUREG-1793 Section 9.2.9 findings and conclusions related to controlling release of radioactive materials. Therefore, RAI-SRP9.3.3-SBPA-02 is resolved.

In DCD Revision 17, Tier 1, Section 2.3.29, the applicant removed the references to the waste water retention basin and replaced them with reference to the turbine building sump. These changes are consistent with the changes to Tier 2 Section 9.2.9.5 (discussed above). In short, since the waste water retention basin is no longer in the scope of the certification, the applicant relocated the system's detection and isolation functions to the turbine building sump, rather than the waste water retention basin, thus providing conformance with GDC 60. Therefore, the staff finds this Tier 1 change acceptable.

9.2.9.3 Conclusion

The staff reviewed the applicant's proposed changes to the AP1000 regarding the nonradioactive WWS as documented in AP1000 DCD Tier 2 Subsections 9.2.9.2.1, 9.2.9.2.2, 9.2.9.5, and 9.2.11.2, and Tier 1 Section 2.3.29. The staff finds that the proposed changes meet the acceptance criteria in NUREG-0800 Section 9.3.3. The staff concludes that the AP1000 WWS continues to meet all applicable acceptance criteria and that proposed changes are properly documented in the updated AP1000 DCD. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii); in that they contribute to increased standardization of the certification information.

9.2.10 Hot Water Heating System

9.2.10.1 Summary of Technical Information

In NUREG-1793 the staff approved Section 9.2.10, “Hot Water Heating System” of the AP1000 DCD, Revision 15.

In AP1000 DCD Revision 17, the applicant identified the following Tier 2 changes associated with the Hot Water Heating System, DCD Sections 9.2.10.2.1, 9.2.10.2.3, and 9.2.10.3.

1. The applicant proposed to modify the AP1000 DCD Section 9.2.10.2.1, “General Description,” to delete the method of matching the system heat load and regulating system temperature using a heater bypass valve.

2. The applicant proposed to modify the AP1000 DCD Section 9.2.10.2.3 “System Operation,” to delete reference to a three-way diverting valve that regulates the temperature of the hot water system. No technical basis for this change is provided. The staff notes that the system operation already includes a description of the intended method temperature control to individual heating coils. Temperature regulation through the use of a heater bypass is not required in order for the system to perform its functions.

3. The applicant proposed to modify the AP1000 DCD Section 9.2.10.3, “Safety Evaluation,” to delete reference to a three-way diverting valve that regulates the temperature of the hot water system.

The applicant identified no Tier 1 changes associated with the hot water heating system.

9.2.10.2 Evaluation

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

The staff reviewed all changes to the hot water heating system (VYS) in the AP1000 DCD Revision 17 and finds them acceptable.

The VYS has no safety-related function and, therefore, no nuclear safety design basis. The following evaluation discusses the results of the staff’s review of the Revision 17 changes.

<u>Deletion of 3-Way Diverting Valve for Temperature Regulation</u>

In DCD Revision 17, Sections 9.2.10.2.1, 9.2.10.2.3, and 9.2.10.3, the applicant proposed to delete discussion of the method of regulating VYS system temperature through the use of a 3-way diverting valve that would bypass the hot water heaters. The applicant deleted this information to allow flexibility in designing and constructing the VYS. The staff finds this change acceptable, since it does not affect the NUREG-1793 Section 9.2.10 findings and conclusions.

9.2.10.3 Conclusion

In NUREG-1793, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 VYS as documented in DCD, Revision 17. The staff concludes that the VYS will continue to comply with the conclusions documented in NUREG-1793 Section 9.2.10. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to increased standardization of the certification information.

9.3 Process Auxiliaries

9.3.1 Compressed and Instrument Air System

9.3.1.1 Summary of Technical Information

In the certified design, AP1000 DCD Revision 15, the staff approved Section 9.3.1, "Compressed and Instrument Air System."

In AP1000 DCD Revision 17 the applicant identified the following Tier 2 changes associated with the compressed and instrument air system (CAS), DCD Table 9.3.1-1. Table 9.3.1-2, Table 9.3.1-3, and Table 9.1.3-4.

1. The applicant proposed to change the AP1000 DCD Section 9.3.6.3.7, Table 9.3.1-1, and Figure 9.3.6-1 Sheet 1 of 2, such that the normal position of air operated containment isolation valve CVS-PL-V092 is open. CVS-PL-V092, "Hydrogen Addition Containment Isolation Valve," is now normally open and fails closed on loss of air. Conforming changes have been made to Table 9.3.1-1, Safety-Related Air-Operated Valves and Figure 9.3.6-1.

2. The applicant proposed to modify the AP1000 DCD Table 9.3.1-2, "Nominal Component Design Data – Instrument Air Subsystem," to clarify that the capacity of the air receivers is a minimum of 19 cubic meters (m^3) (672 cubic feet (f^3)) instead of exactly 19 m^3 (672 f^3). The technical basis for this change is to account for growth in demand on the system by ensuring that the receiver tank capacity is at least 19 m^3 (672 f^3).

3. The applicant proposed to modify the AP1000 DCD Table 9.3.1-3, "Nominal Component Design Data – Service Air Subsystem," to clarify that the capacity for the air receiver is a minimum of 19 m^3 (672 f^3) instead of exactly 19 m^3 (672 f^3). The technical basis for this change is to account for growth in demand on the system by ensuring that the receiver tank capacity is at least 19 m^3 (672 f^3).

4. The applicant proposed to modify the AP1000 DCD Table 9.3.1-4, "Nominal Component Design Data – High Pressure Air Subsystem," to clarify that the system design pressure is reduced from 34.5 MPa (5,000) psig to 27.6 MPa (4,000 psig). The technical basis for this change is to revise the system design pressure.

9.3.1.2 Evaluation

The staff reviewed all changes to the CAS in the AP1000 DCD Revision 17 in accordance with NUREG-0800 Section 9.3.1, "Compressed Air System." The regulatory basis for Section 9.3.1 of the AP1000 DCD is documented in NUREG-1793, which states that staff acceptance of the design is contingent on compliance with the following requirements in NUREG-0800 Section 9.3.1:

- GDC 1, as it relates to systems and components being designed, fabricated, and tested to quality standards in accordance with the importance of the safety functions to be performed.

- GDC 2, as it relates to the capability of safety-related CAS components to withstand the effects of earthquakes.

- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions

The CAS has no safety-related function other than containment isolation. The following evaluation discusses the results of the staff's review of the Revision 17 changes.

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

<u>Containment Isolation Valve CVS-PL-V092 Change in Normal Position</u>

In AP1000 DCD Table 9.3.1-1, the applicant proposed to change the normal position of air-operated containment isolation valve, CVS-PL-V092, from closed to open. The applicant made a conforming change on Figure 9.3.6-1 Sheet 1 of 2, "Chemical and Volume Control System Piping and Instrumentation Diagram," showing valve CVS-PL-V092 to be normally open, in order to facilitate zinc acetate injection into the RCS. The evaluation of the valve position is in Section 9.3.6 of this report. The basis for this proposed change is APP-GW-GLN-002 (TR-32).

Based on its evaluation, the staff finds that this change does not affect the NUREG-1793 Section 9.3.1 assumptions, findings, or conclusions with respect to compliance with GDC 1, 2, or 5, as referenced in NUREG-0800 Section 9.3.1, and is, therefore, acceptable.

The impact of this change in the containment isolation system is discussed in Section 6.2.4, "Containment Isolation System," of this report.

<u>DCD Tables 9.3.1-2 and 9.3.1-3, Nominal Component Design Data – Instrument Air and Service Air Subsystems</u>

In DCD Tables 9.3.1-2 and 9.3.1-3, the applicant clarified that the capacity of the air receivers is a minimum of 19 m^3 (672 f^3) instead of being exactly 19 m^3 (672 f^3). This change is conservative in that the receiver tank capacity can be greater than 19 m^3 (672 f^3) and still meet design criteria.

Based on its evaluation, the staff finds that this change does not affect the NUREG-1793 Section 9.3.1 assumptions, findings, or conclusions with respect to compliance with GDC 1, 2, or 5, as referenced in NUREG-0800 Section 9.3.1, and is, therefore, acceptable.

<u>DCD Table 9.3.1-4, Nominal Component Design Data – High Pressure Air Subsystem</u>

In DCD Table 9.3.1-4, the applicant modified the high-pressure air subsystem design pressure from 34.5 MPa (5,000) psig to 27.6 MPa (4,000 psig).

The staff finds that this change does not affect the NUREG-1793 Section 9.3.1 assumptions, findings, or conclusions with respect to compliance with GDC 1, 2, or 5, as referenced in NUREG-0800 Section 9.3.1, and is, therefore, acceptable.

9.3.1.3 Conclusion

In NUREG 1793 and Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 CAS as documented in DCD, Revision 17. The staff concludes that the CAS will continue to comply with GDC 1, 2, and 5 as stated in NUREG-1793 Section 9.3.1. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to increased standardization of the certification information.

9.3.2 Plant Gas System

In the certified design, AP1000 DCD Revision 15, the staff approved Section 9.3.2, "Plant Gas System."

In DCD Revision 17, the applicant proposed to make the following changes to DCD Revision 15:

1. In Section 9.3.2.2.1, "General Description" the applicant proposed to change the location of both the packaged nitrogen system and the carbon dioxide portion of the plant gas system from inside the turbine building to "in the gas storage area in the yard."

2. In Section 9.3.2.2.2, "Component Description" the applicant proposed to change the cryogenic liquid carbon dioxide insulated storage tank from double wall to single wall.

Section 3.5.1.4 of this report includes an analysis of storage tanks as a potential missile source and Section 6.4 of this report includes analyses of onsite chemicals.

9.3.3 Primary Sampling System

9.3.3.1 Summary of Technical Information

In the certified design, AP1000 DCD Revision 15, the staff approved Section 9.3.3, "Primary Sampling System."

In DCD Revision 17, the applicant proposed to make the following change to DCD Revision 15:

- In Section 9.3.3.2.2, "Nuclear Sampling System - Gaseous" the applicant changed the discharge location of the purge gas return from the effluent holdup tank of the liquid radwaste system to the containment sump.

9.3.3.2 Evaluation

The staff reviewed the proposed change to Tier 2 Section 9.3.3.2.2 of the AP1000 DCD Revision 17, in accordance with the guidance of NUREG-0800 Sections 9.3.2, 9.3.3, and 11.5. The regulatory basis for Section 9.3.3 of the AP1000 DCD is documented in Section 9.3.3 of NUREG-1793.

The specific criterion that applies to the proposed DCD change is 10 CFR 52.63(a)(1)(vii) which concerns contribution to increased standardization of the certification information.

NUREG-0800 Section 9.3.2 states that provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA. As the sample line for the containment atmosphere returns the purge gas back to the containment, the proposed change meets the acceptance criteria of the NUREG-0800 guidance and the ALARA requirements. The staff finds the change in location of the purge line discharge from the effluent holdup tank to the containment sump acceptable.

9.3.3.3 Conclusion

The staff reviewed the applicant's proposed changes to the AP1000 regarding the Primary Sampling System as documented in AP1000 DCD Tier 2 Section 9.3.3, Revision 17. The staff concludes that the AP1000 Primary Sampling System continues to meet all applicable acceptance criteria and the proposed changes are properly documented in the updated AP1000 DCD. The proposed change meets the criterion in 10 CFR 52.63(a)(1)(vii) on the basis that it contributes to increased standardization of the certification information and is, therefore, acceptable.

9.3.5 Equipment and Floor Drainage System

9.3.5.1 Summary of Technical Information

Section 9.3.5, "Equipment and Floor Drainage System" (EFDS) of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant proposed to make changes to Sections 9.3.5.1.2, "Power Generation Design Basis," and 9.3.5.2.2, "Component Description" of the certified design as follows:

1. The applicant clarified that an exception exists to the minimum slope of 10.42 millimeter per meter (mm/m) (1/8 inches per foot (in/ft)) for drain lines, which is the embedded drain piping in area 2 of the auxiliary building at elevation 20.27 m (66 ft, 6 in) with a minimum slope of 5.21 mm/m (1/16 in/ft) for embedded drain piping.

2. The applicant clarified that each sump is fitted with a vent connection to the radiologically controlled area ventilation system (VAS) exhaust system to exhaust potential sump gases, instead of exhausting into the room.

9.3.5.2 Evaluation

The staff reviewed the proposed changes to Tier 2 Sections 9.3.5.1.2 and 9.3.5.2.2 of the AP1000 DCD Revision 17, in accordance with the guidance of NUREG-0800 Section 9.3.3.

The regulatory basis for Section 9.3.5 of the AP1000 DCD is documented in Section 9.3.5 of NUREG-1793.

The specific criterion that applies to the proposed DCD changes is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

DCD Revision 15, Section 9.3.5.1.2 provides for a minimum slope of 10.42 mm/m (1/8 in/ft) for embedded drain lines in Level 1 (Area 2) of the auxiliary building at elevation 20.27 m (66 ft, 6 in). In DCD Revision 17, Section 9.3.5.1.2, the applicant changed the minimum slope from 10.42 mm/m (1/8 in/ft) to 5.21 mm/m (1/16 in/ft) for embedded drain lines.

In order for the staff to complete its evaluation, the staff asked the applicant, in RAI-SRP9.3.5-SBPA-01, to justify the change from the minimum slope of 10.42 mm/m (1/8 in/ft) of drain pipe length, and to address its impact on flooding in that part of the auxiliary building by evaluation.

The applicant provided a response to RAI-SRP9.3.5-SBPA-01 in a letter dated September 17, 2009. The RAI response stated that the drain lines located in Level 1 of the auxiliary building (Elevation 20.3 m (66 ft, 6 in)), nonradioactive controlled area, are not credited in the flooding analysis. Because no credit is taken for the drains in this location, the staff finds the applicant's justification for the embedded drain piping slope in this area acceptable and, therefore, GDC 4 is met. Therefore, RAI-SRP9.3.5-SBPA-01 is resolved.

In DCD Section 9.3.5.2.2, the applicant clarified that each sump is fitted with a vent connection to the VAS exhaust system instead of exhausting directly into the room. This allows potential sump gases from each sump to be directed to an exhaust system for the control of airborne radioactivity. The staff finds that this change would minimize potential release of airborne radioactivity or other harmful gases in sump rooms and is, therefore, acceptable.

The staff finds that the applicant adequately identified changes that would not adversely impact the compliance of the EFDS with the guidance in NUREG-0800 Section 9.3.3. Thus, the staff finds that the applicant continues to meet GDC 2, 4, and 60.

9.3.5.3 Conclusion

In NUREG 1793, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 EFDS as documented in DCD, Revision 17. The staff concludes that the EFDS will continue to comply with the conclusions documented in NUREG-1793 Section 9.3.5. The proposed change meets the criterion of 10 CFR 52.63(a)(1)(vii); on the basis that it contributes to increased standardization of the certification information.

9.3.6 Chemical and Volume Control System

9.3.6.1 Summary of Technical Information

In a letter dated April 5, 2006, the applicant submitted TR-32, “Zinc Addition,” APP-GW-GLN-002, which proposed changes to the AP1000 CVS design to incorporate the ability to inject a small quantity of zinc acetate into the RCS through the CVS. In TR-80, “Markup of AP1000 Design Control Document Chapter 7,” APP-GW-GLR-080, Revision 0, of October 2007, the applicant proposed CVS design changes related to boron dilution events. In DCD Revision 17, the applicant proposed to change DCD Section 9.3.6, “Chemical and Volume Control System,” to incorporate the CVS design changes described in TR-32 and TR-80.

9.3.6.2 Evaluation

The AP1000 CVS as described in DCD Section 9.3.6 is designed to perform the functions of the RCS purification, inventory control and makeup, chemical shim and control, oxygen control, pressurizer auxiliary spray, and borated makeup to the auxiliary equipment. The safety evaluation accepting the CVS design of DCD Section 9.3.6, Revision 15, was described in NUREG-1793, Section 9.3.6. The review was performed using the guidance of NUREG-0800 Section 9.3.4, “Chemical and Volume Control System (PWR) (Including Boron Recovery System),” to assess compliance with the requirements for system performance of necessary functions during normal, abnormal, and accident conditions described in GDC 1; GDC 2; GDC 5; GDC 14, “Reactor Coolant Pressure Boundary”; GDC 29, “Protection Against Anticipated Operational Occurrences”; GDC 33, “Reactor Coolant Makeup”; GDC 35, “Emergency Core Cooling”; GDC 60; and GDC 61.

In Revision 17 of DCD, the applicant proposed changes to DCD Section 9.3.6 on the CVS design with: (1) the capability for zinc addition to the RCS; and (2) modifications related to boron dilution events. The staff’s review of these changes is to assure continued compliance with the relevant requirements specified in the above GDCs.

The specific criterion that applies to the changes referred to above include 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

<u>Zinc Addition</u>

In TR-32, the applicant proposed a modification to the CVS design to provide the capability to inject a small quantity of zinc acetate into the RCS. In DCD Revision 17, the following subsection is added to DCD Section 9.3.6.2 as one of the CVS chemical control functions:

> 9.3.6.2.3.3 Zinc Addition
>
> A soluble zinc compound may be added to the coolant as a means to reduce radiation fields within the primary system and to reduce the potential for crud-induced power shift (CIPS). The zinc used may be either natural zinc or zinc depleted of ^{64}Zn.

Also, DCD Section 9.3.6.2.1.1, “Ionic Purification,” is revised to include the removal of zinc during periods of zinc addition as an added ionic purification function of the mixed bed demineralizers in the purification loop, in addition to removing ionic corrosion products and certain ionic fission products.

TR-32 provides a description and evaluation of the proposed CVS design change to incorporate the ability to inject a small quantity of zinc acetate into the RCS through the CVS. For AP1000, zinc addition will be an optional mode of operation, and the equipment specifically used for storing and pressurizing the zinc acetate is not described. However, as discussed below, minor changes to the base AP1000 CVS design are required to allow for zinc addition to be used as an optional mode of operation.

TR-32 also states that zinc acetate will be added using the same piping and valving as the hydrogen (H_2) addition. The proposed hardware change is to replace a portion of the one-inch pipe downstream of the containment isolation valve with a heavier wall half-inch pipe. To accomplish this design change, in DCD Revision 17, Figure 9.3.6-1 was changed by (1) changing the hydrogen addition line from "H2 ADD" to a hydrogen/zinc addition line "H2/ZINC ADD"; (2) adding a 2.5 cm by 1.27 cm (1 in by .5 in) reducer downstream of valve V065 in the "H2/ZINC ADD" line; and (3) renumbering the portion of the "H2/ZINC ADD" line downstream of the reducer L064 with the specification changed from 2.5 cm (1 in) BBC to 1.27 cm (0.5 in) BBC. According to TR-32, this is made to reduce the piping volume and reduce the transit time for the H_2 and the zinc acetate supply, and will not alter the load of the supply piping. The staff finds this change acceptable because there is no effect on the integrity of the reactor coolant pressure boundary. Also, in DCD Revision 17, in Section 9.3.6.3.7, the hydrogen addition containment isolation valve V092, which is located outside the containment, was changed from the "normally closed, fail closed" position to "normally open, fail closed" position; and V092 in Figure 9.3.6-1 was changed to the "normally open" position. According to TR-32, this change is made to reflect the time that the valve must be open to permit zinc additions. The staff finds this change acceptable because this containment isolation valve automatically closes on a containment isolation signal from the protection and safety monitoring system and, therefore, the containment isolation function is not affected.

Chemistry and Materials Impacts During Normal Operation

According to TR-32, zinc addition has been demonstrated to change the oxide film on primary piping and components, significantly reducing occupational exposure (due to less nickel and cobalt in deposit) and the potential for CIPS. The applicant also stated that laboratory tests indicate a beneficial effect of zinc addition on the major materials in a PWR system because of the reduced corrosion rates and that operating industry experience has shown that up to 40 parts per billion can be added with no adverse effects. The applicant stated that the effect of zinc on the reactor coolant was calculated to be less than 0.2 pH units, which it considers negligible. The staff reviewed industry experience with zinc injection in operating plants and agrees that there is sufficient experience to support the conclusion that there is no deleterious effects on reactor coolant chemistry or reactor coolant pressure boundary materials at zinc injection concentrations up to the maximum limit proposed by the applicant. Although the beneficial effect of zinc addition on major materials at higher concentrations has not yet been fully established through laboratory tests, the reduction in occupational exposure and CIPS has been demonstrated through operating experience.

Post Accident Water Chemistry Impacts

Hydrogen generation caused by corrosion of reactive metals such as zinc and aluminum following a design-basis accident (DBA) is a concern that should be addressed according to

NUREG-0800 Section 6.1.1. However, according to TR-32, because zinc exists as a divalent cation (Zn^{+2}) in solution in the primary coolant and embedded in the corrosion film, hydrogen generation is not expected. The staff agrees that because zinc is added as zinc acetate salt rather than metallic zinc, the zinc would only exist in ionic form that cannot produce hydrogen as a corrosion reaction byproduct. Further, even if the zinc cations in solution could react to form hydrogen, the staff concurs that the amount of zinc is small and would not produce a significant amount of hydrogen.

In addition, given that the volume of reactor coolant is small compared to the borated water volume in the sump, the applicant stated that the effect of zinc on the sump pH following a DBA would also be negligible. The staff finds this statement reasonable due to the low concentration of zinc compared to the concentrations of buffer and boric acid in the sump water following a DBA.

Based on the above, the staff finds the applicant's evaluation of zinc addition on post accident chemistry to be acceptable because the evaluation is based on basic principles of chemistry and the quantity of zinc involved is small.

Fuel Corrosion and Crud Effects

According to TR-32, the addition of zinc to the RCS could result in additional crud deposit on the fuel cladding surface. The applicant performed oxide thickness measurements that showed the crud deposit was thin and the effect of the corrosion rate with zinc injection was statistically insignificant as compared to the corrosion rate without zinc injection. In response to RAI-TR32-07, the applicant provided data from 12 fuel surveillance campaigns, based on which it concluded that there were no observed adverse effects on the fuel cladding performance due to zinc addition. The staff reviewed the plant surveillance data for oxide thickness that were provided in the response to RAI-TR32-07 and agrees with the applicant's conclusion because there was no statistical difference in the measured oxide thickness for cladding with and without exposure to zinc in the coolant. Further, the staff's review of industry experience related to zinc impacts on fuel indicates there are no adverse effects for low- to medium-duty cores. The applicant indicated that the absence of deleterious effects on cladding will be confirmed through cycle-specific reload analyses with zinc addition. The staff identified Open Item OI-SRP9.3.6-SRSB-01 for the applicant to explain how cycle-specific reload analyses can confirm no adverse effect of zinc addition for the AP1000 with a high-duty core and why a fuel surveillance program is not needed to confirm the absence of adverse crud effects.

To provide further definition to assist the applicant in understanding the open item, the staff communicated the following questions to the applicant via an email dated October 23, 2009:

1. Is the AP1000 core design as described in DCD Revision 17 considered a high-duty core?

2. If the AP1000 is considered a high-duty core design, how will it be assured that there will not be problems with excessive crud buildup or uneven crud buildup in operating AP1000 plants? This answer may necessitate either operating experience that demonstrates that a fuel surveillance program is unnecessary, or a recommendation for a fuel surveillance program to be implemented (COL Item).

3. Explain how cycle-specific reload analyses would confirm that zinc would not have a significant effect on cladding corrosion.

4. Since one of the objectives of zinc addition is to reduce the potential for CIPS, will an evaluation of CIPS potential be performed as part of the cycle-specific reload analysis? If so, provide details of the evaluation and a relative comparison to operating plants with and without CIPS. Also, since zinc injection will initially increase the reactor coolant Ni concentration, explain in detail the zinc injection strategy to be employed to minimize the CIPS potential.

In a letter dated February 18, 2010, the applicant responded to Open Item OI-SRP9.3.6-SRSB-01 and specifically the 4 questions as follows:

1. The applicant stated that “According to EPRI HDCI [High-duty Core Index], AP1000 would be classified as a low to medium duty plant.” The applicant provided a table containing the parameters used to calculate the HDCI. The staff performed a confirmatory calculation using the same parameters and obtained the same result. The applicant further stated that “since in terms of boiling duty the AP1000 is approaching that of other Westinghouse high duty plants, we are conservatively treating AP1000 as a high duty plant. There are other currently operating PWRs that are higher duty and also use zinc addition, so AP1000 is bounded by current operating experience.”

The staff finds the response to Item 1 acceptable because a confirmatory calculation verified the HDCI for the AP1000, and because the applicant is conservatively treating the AP1000 core design as high-duty.

2. The applicant indicated that zinc addition will be employed to reduce RCS surface corrosion rates beginning from hot functional testing. The applicant also stated that experience with zinc addition in current PWRs following steam generator replacement indicates substantial benefits in reducing corrosion rates when zinc is applied to fresh metal surfaces such as those found following steam generator replacement. Papers on first cores show similar benefits will occur with AP1000, but right from the beginning of plant operation.

   The applicant also indicated that there have been several additional high-duty plants that began zinc addition since 2003, which were not all reflected in the EPRI Zinc Addition Guidelines cited by the staff. The applicant listed Callaway, Vogtle 1 and 2, Byron 2, Braidwood 2, South Texas 1 and 2, and Watts Bar 1 as high-duty plants that have successfully operated with zinc addition and no problems with crud deposition or fuel performance related to zinc. The applicant also stated that fuel examinations following zinc addition have been completed at numerous high-duty PWRs and continue to show no increase in cladding corrosion and no deleterious impact on fuel crud deposits.

   Finally the applicant stated that AP1000 will have a robust fuel inspection program looking not only at crud but other things using EPRI fuel reliability guidelines.

The staff finds the applicant’s response to Item 2 acceptable because:

a) Based on our review of industry experience related to zinc addition, the staff agrees that excessive nickel release should not occur if zinc addition starts during hot functional testing. This conclusion is supported by data from plants that started injecting zinc concurrently with steam generator replacement reported in EPRI

Report #1013420, “Pressurized Water Reactor Primary Water Zinc Application Guidelines,” of December 2006.

b) The applicant cited additional industry experience with zinc addition in high-duty cores that supports the applicant’s assertion that zinc addition will not cause increased CIPS risk in high-duty cores.

3. The applicant stated that zinc has been shown not to interact with zircaloy clad fuel and does not increase clad corrosion. The applicant further stated that reload-specific corrosion analyses do not need to be penalized due to zinc addition because as zinc began to be applied to higher duty cores additional fuel surveillance was undertaken to determine if any increased cladding corrosion was occurring. Finally the applicant stated that the surveillances have not shown any indication of enhanced clad corrosion for zinc application in these higher duty cores where crud deposits were present.

   Additionally, the applicant indicated that the cycle specific reload analysis described in the response to Item 4 will demonstrate that the reload designs should not result in excessively thick crud deposits.

The staff finds the applicant’s response to Item 3 acceptable because its review of operating experience related to zinc addition confirms the applicant’s claim that there are no adverse effects on fuel-cladding corrosion caused by zinc addition.

4. In response to the first part of Item 4, the applicant stated that a CIPS risk analysis is currently performed using EPRI guidelines and methods for every reload design performed in Westinghouse, and that this process will be performed as part of the initial core and each reload core analysis for AP1000. The applicant further stated that the VIPRE/BOA methods will be used as recommended in the EPRI AOA Guidelines (PWR Axial Offset Anomaly (AOA) Guidelines, Revision, 1008102, Final Report, June 2004, EPRI).

   In response to the second part of Item 4, the applicant indicated that no increase in reactor coolant nickel concentration is expected since zinc addition will start during hot functional testing; thus, the zinc will be incorporated into the corrosion films as they form on the fresh metal surfaces. According to the applicant, this is a more favorable situation compared to the addition of zinc to existing plants with mature corrosion films, which causes nickel to be displaced from the corrosion films. Therefore, the applicant’s response indicates they expect no increase in crud due to displacement of nickel.

The staff finds the response to the first part of Item 4 acceptable because the applicant will be using accepted industry computer codes for analysis of the crud and CIPS risk (VIPRE/BOA). This is consistent with the approach used for operating Westinghouse plants.

The staff finds the response to the second part of Item 4 acceptable because, based on its review of industry experience related to zinc addition; the staff agrees that excessive nickel release should not occur if zinc addition starts during hot functional testing. This conclusion is supported by data from plants that started injecting zinc concurrently with steam generator replacement reported in EPRI Report #1013420.

The staff finds that the applicant has adequately addressed potential fuel corrosion and crud effects of zinc addition because:

- The applicant has presented sufficient industry experience with zinc addition in reactors with high-duty cores to demonstrate that problems with cladding corrosion, excessive crud buildup, or CIPS are not expected.

- The applicant has proposed to use industry-accepted computer codes to model and predict crud formation, for each reload, to confirm that a problem will not occur with crud or CIPS.

Although the applicant did not propose to modify the DCD to include any of the information supplied in the open item response, or provide a COL information item to ensure the COL performs the activities described, the staff notes that there is no specific regulatory requirement for a plant to monitor, model or test for crud buildup or CIPS. Therefore, the proposed reload analyses and fuel surveillance program are not mandatory. The staff considers Open Item OI-SRP9.3.6-SRSB-01 resolved.

Based on the above evaluation, the staff concludes that the zinc addition into the RCS as an operational option, and the associated DCD changes associated with zinc addition discussed above, are acceptable.

<u>Modification to Boron Dilution Event</u>

In DCD Revision 17, the applicant proposed the following changes to DCD Section 9.3.6 associated with boron dilution events:

In DCD Revision 17, Section 9.3.6.3.7, “Chemical and Volume Control Systems Valves,” a sentence is added regarding the makeup line containment isolation valves, which are normally open motor-operated globe valves, to state that the valves close on a source range flux doubling signal to terminate possible unplanned boron dilution events. In DCD Section 9.3.6.7, “Instrumentation Requirements,” a change is also made to the “makeup isolation valves” to state that the two makeup isolation valves automatically close on a signal from the protection and safety monitoring system derived from source range doubling high-2 pressurizer level, high steam generator level, or a safeguards signal coincident with high-1 pressurizer level. In DCD Revision 17, Section 9.3.6.4.5.1, “Boron Dilution Events,” the CVS response to a boron dilution event is revised from “closing either one of two redundant safety-related, air-operated valves from the demineralized water system to the makeup pump suction” to “closing redundant safety-related valves, tripping the makeup pumps and/or aligning the suction of the makeup pumps to the boric acid tank.” The description of dilution events during shutdown is also revised to state that “the source range flux doubling signal is used to isolate the makeup line to the RCS by closing the two safety-related, MOVs, isolate the line from the demineralized water system by closing the two safety-related, air-operated valves and trip the makeup pumps.” This is a change from the statement in Revision 15 that “the source range flux doubling signal is used to isolate the line from the demineralized water system by closing the two safety-related, remotely operated valves. The three-way pump suction control valve aligns the makeup pumps to take suction from the boric acid tank and therefore stops the dilution.”

In TR-80, Item II.9 “Flux Doubling/Boron Dilution Modifications,” provides a rationale for the changes related to boron dilution events in DCD Section 9.3.6. The existing CVS design realigns the makeup pump suction from the demineralized water tank to the boric acid tank to terminate the potential boron dilution and to begin to reborate the RCS to restore shutdown margin. These actions would initially cause the boron dilution to continue because the volume of water in the makeup line path would still be unborated until borated water from the boric acid tank begin to reach the RCS. The function is therefore changed to close the makeup line isolation valves (as well as the demineralized water isolation valves) to terminate the event as soon as possible. Long term recovery from the event would then be accomplished using either a different flow path with a smaller unpurged volume or by using the makeup line after purging most of the unborated water in it. The staff finds that the revised boron dilution events description is consistent with the modifications on the CVS makeup isolation valves in DCD Sections 9.3.6.3.7 and 9.3.6.7. This is also consistent with the boron dilution events described in Section 15.4.6.1 and Sections 15.4.6.2.2 through 15.4.6.2.5 for boron dilution events during Modes 5 through mode 2, respectively. Therefore, the staff finds the changes in the AP1000 DCD Sections 9.3.6.3.7, 9.3.6.4.5.1, and 9.3.6.7 are acceptable.

In DCD Section 9.3.6.6.1.2, “Flow Testing,” the maximum makeup flow for the pump testing verification is changed from 757 Lpm to 662 Lpm (200 gpm to 175 gpm) with both pump operating. The change in the maximum makeup flow rate from 757 Lpm to 662 Lpm (200 gpm to 175 gpm) reflects the maximum flow rate through the cavitating venturi nozzle on the makeup pump discharge header. The CVS makeup pump testing is performed to verify that the maximum makeup flow with both makeup pumps operating is less than 662 Lpm (175 gpm). The staff finds that this maximum makeup flow rate is consistent with the assumptions of the boron dilution events analyzed in DCD Revision 17, Section 15.4.6, where the assumed unborated water flow rate is also changed from 757 Lpm to 662 Lpm (200 gpm to 175 gpm). The reduction of the maximum makeup flow to 662 Lpm (175 gpm) limits the unborated water flow rate to 662 Lpm (175 gpm) in the inadvertent boron dilution events and, thus, allows more time for isolation of the unborated water and termination of the events. Therefore, the staff finds that this change is acceptable.

In DCD Revision 17, Table 9.3.6-2, the following CVS nominal design parameters are changed: the letdown HX shell side and tube side design temperatures are changed from 93.3 °C (200 °F) and 343.3 °C (650 °F), respectively, to 65.5 °C (150 °F) and 315.5 °C (600 °F); the design flows of the mixed bed and cation bed demineralizers, and the reactor coolant filter are changed from 379 Lpm to 946 Lpm (100 gpm to 250 gpm); and the boric acid storage tank volume is changed from 264,979 L to 278,28 5 L (70,000 to 73,515 gallons). In RAI-SRP9.3.6-SRSB-02 the staff requested that the applicant provide the basis and justification for these changes to determine their acceptability. In its response dated January 26, 2009, the applicant provided the bases for these changes. The applicant states that the shell side temperature of the letdown HX cannot exceed 65.5 °C (150 °F) because shell side coolant is the CCS with a normal operating temperature of no higher than 37.8 °C (100 °F). The tube side temperature cannot exceed 315.5 °C (600 °F) as the purification flow is first cooled through the regenerative HX. The staff finds the tube side temperature acceptable because the purification flow is drawn from the RCS cold leg, which has a temperature lower than 315.5 °C (600 °F). The staff concludes that the changes of letdown HX shell side and tube side temperatures to 65.5 °C (150 °F) and 315.5 °C (600 °F), respectively, still maintain sufficient margin to the operating temperatures and are, therefore, acceptable.

The applicant states that the changes of the design flows from 379 Lpm to 946 Lpm (100 gpm to 250 gpm) for the demineralizers and the reactor coolant filters accommodate shutdown purification flows that could be as high as 810 Lpm (214 gpm) when the normal RNS provides the motive force for reactor coolant purification. The staff considers these design flow increases conservative changes that provide margin for the shutdown purification flow and are, therefore, acceptable. The applicant states that the boric acid storage tank increase in volume from 264,979 L to 278,285 L (70,000 to 73,515 gallons) represents the usable volume of the tank, which includes the volume to accommodate a shutdown to cold shutdown followed by refueling at the end of the fuel cycle, plus the volume needed for normal operation and operating margin, and this increased volume is calculated with updated inputs that more accurately represents the AP1000 design. Because this more accurately reflects the design information, the staff finds this change acceptable.

In DCD Revision 15, the CVS demineralizer resin flush line containment isolation thermal relief valve (CVS-PL-V042) was located outside of the containment and discharged to the WLS waste holdup tank. In DCD Revision 17, Figure 9.3.6-1 is revised to relocate CVS-PL-V042 to inside containment between the two containment isolation valves. This relief valve is provided to prevent overpressurization of the resin sluice line that is used to sluice resin from the mixed bed and cation bed demineralizers to the waste processing system. The staff reviewed this change and finds that the location of the relief valve inside containment does not affect the functional capability of the CVS, but provides a protection against release of potential radioactive products outside containment. Therefore, the staff concludes that the proposed change is acceptable.

9.3.6.3 Conclusion

The staff reviewed the changes to DCD Section 9.3.6 regarding the AP1000 CVS design as described in DCD Revision 17. Based on the evaluation described above, the staff concludes that these changes would not adversely impact the required AP1000 CVS functions, and that the requirements of GDC 1, 2, 5, 14, 29, 33, 35, 60, and 61 continue to be met. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii); in that they contribute to increased standardization of the certification information.

9.4 Air Conditioning, Heating, Cooling, and Ventilation System

In NUREG-1793, the staff approved Section 9.4, “Air Conditioning, Heating, Cooling, and Ventilation System,” of AP1000 DCD, Revision 15. In AP1000 DCD, Revision 17, the applicant proposed changes to this section, supported by TR-103.

The staff reviewed the changes to AP1000 DCD, Revision 15, Section 9.4, which are described in Revision 17. NUREG-1793 includes the regulatory basis for Section 9.4 of AP1000 DCD. The staff reviewed the proposed changes to DCD Section 9.4 against the applicable acceptance criteria in NUREG-0800 related to Section 9.4. Those changes that involve NRC review considerations as reflected in NUREG-0800 are described and evaluated in this section.

The specific criterion that applies to the changes evaluated in this section is 10 CFR 52.63(a)(1)(vii), which concerns the contribution to increased standardization of the certification information.

9.4.1 Nuclear Island Nonradioactive Ventilation System (VBS)

9.4.1.1 Summary of Technical Information

In Section 9.4.1.2.2, “Component Description,” the applicant proposed to change the humidifier description such that the design and rating of the VBS Humidifiers will be in accordance with ARI 640, “Commercial and Industrial Humidifiers.”

The DCD previously referenced standard ARI 620, “Self Contained Humidifiers for Residential Applications” in the design of the humidifiers. This was incorrect as ARI 620 states that the intended application of the humidifiers is typically for non-ducted applications, and is independent of a central air system. The AP1000 uses humidifiers in ducted central air applications; therefore, ARI 640, “Commercial and Industrial Humidifiers” is the correct specification, and has replaced ARI 620. This correction applies to VAS, VBS, health physics and hot machine shop heating, ventilation, and air conditioning (HVAC) system (VHS), turbine building ventilation system (VTS), and annex/auxiliary buildings nonradioactive HVAC system (VXS) systems.

In Section 9.4.1.2.3.1, “Main Control Room/Control Support Area HVAC Subsystem,” the applicant proposed various modifications to both the normal plant operation and abnormal plant operation sections.

In NUREG-1793, the staff approved a VBS HVAC system that has one heater in each air handler and one pair of temperature sensors to control the temperature in the control support area (CSA) and MCR areas. The airflow to each space is selected to properly cool each space at the summer design weather conditions. During winter conditions, cooling is required to maintain design conditions in some spaces, including the MCR, some electric/electronic equipment spaces, and the CSA computer rooms. Since the VBS system, as previously configured, could not heat some spaces and cool others simultaneously, additional heaters and temperature sensors have been added in the return ducting from the computer room for temperature control.

In Table 9.4-1 and Table 9.4.1-1, changes to the VBS leakage rates to MCR/CSA were proposed. The control logic depicted on the Figure 9.4.1-1 (sheet 4 of 7) of the DCD for the VBS fans serving the Class 1E Division B and D electrical rooms has been changed so that starting an air handling unit will start the chilled water system associated with that air handling unit. Previously, starting air handling unit MS03B sent a signal to the VWS to start chilled water pump MP02, which provides chilled water to air handling unit MS03D. In the same manner, starting air handling unit MS03D sent a signal to the VWS to start chilled water pump MP03, which provides chilled water to air handling unit MS03B. In both cases starting the air handling unit fails to start the correct water chiller and, therefore, the cooling system would not have operated correctly. The logic has been corrected so that starting air handling unit MS03B will start chilled water pump MP03 and starting air handling unit MS03D will start chilled water pump MP02; thus, starting each air handling unit will start the respective supporting pump.

9.4.1.2 Evaluation

The above mentioned DCD changes are technical improvements, corrections to design errors, or changes of facility descriptions. For the reasons discussed above, all of the changes are acceptable.

9.4.1.3 Conclusion

The NRC staff has reviewed the proposed changes to the VBS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VBS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.2 Annex/Auxiliary Buildings Nonradioactive HVAC System (VXS)

9.4.2.1 Summary of Technical Information

In Section 9.4.2.1.2, “Power Generation Design Basis,” for the annex building nonradioactive HVAC system, there were some room rearrangements: Office areas, conference rooms, and security rooms and areas were added. A central alarm station and a security access area were deleted from the system and a security room in the mechanical equipment room was added.

In Section 9.4.2.2.1.1, “General Area HVAC Subsystem,” the “VXS General Area HVAC Subsystem” was expanded. This expansion would add two more supply air handling units and other equipment to provide ventilated air to personnel areas in the annex building outside the security area.

In Section 9.4.2.2.2, “Component Description,” the applicant changed the humidifier description such that the performance rating of the VAS Humidifiers will be in accordance with ARI 640.

9.4.2.2 Evaluation

The above mentioned DCD changes are technical improvements or changes of facility descriptions. For the reasons discussed above, all of the changes are acceptable.

9.4.2.3 Conclusion

The NRC staff has reviewed the proposed changes to the VXS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VXS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.3 Radiological Controlled Area Ventilation System (VAS)

9.4.3.1 Summary of Technical Information

In Section 9.4.3.2.2, “Component Description,” the applicant proposed to change the humidifier description such that the performance rating of the VAS humidifiers would be in accordance with ARI 640 and Figure 9.4.3-1 would be revised accordingly.

9.4.3.2 Evaluation

The change to ARI 640 is evaluated in Section 9.4.1.

9.4.3.3 Conclusion

The NRC staff has reviewed the proposed changes to the VAS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VAS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.7 Containment Air Filtration System (VFS)

9.4.7.1 Summary of Technical Information

In Section 9.4.7.1.2, the applicant proposed to specify a VFS design pressure of "< -0.125 in. water gauge."

9.4.7.2 Evaluation

This change is evaluated in Section 16.4.12 of this report.

9.4.8 Radwaste Building HVAC System (VRS)

9.4.8.1 Summary of Technical Information

In Section 9.4.8.1.2, "Power Generation Design Basis," the applicant proposed to add "Truck Staging Area" to the rooms/areas covered by the VRS with a specified temperature range.

9.4.8.2 Evaluation

The above mentioned DCD change is an acceptable design improvement.

9.4.8.3 Conclusion

The NRC staff has reviewed the proposed changes to the VRS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VRS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.9 Turbine Building Ventilation System (VTS)

9.4.9.1 Summary of Technical Information

In Section 9.4.9.2.1.2, "Electrical Equipment and Personnel Work Area HVAC," the applicant proposed to provide details regarding the electrical equipment, south bay equipment, and personnel work area air conditioning subsystems including area temperature re-designation from 10.0 °C to 40.5 °C (50 °F to 105 °F) to 10.0 °C to 37.7 °C (50 °F to 100 °F). The following descriptive paragraph was also provided:

> The south bay equipment area HVAC system consists of two 50-percent capacity air handling units of about 7000 cfm capacity each. The air handling units are located on elevation 117'-6" of the turbine building between column 11 and 11.2. The temperature of the room is maintained by the thermostats that control the chilled water control valves for cooling and the integral face bypass dampers for heating. Outside air is mixed with the recirculation air to maintain a positive pressure.

In Section 9.4.9.2.2, "Component Description," the applicant proposed to change the humidifier description such that the design and rating of the VTS humidifiers will be in accordance with ARI 640.

9.4.9.2 Evaluation

The temperature change is acceptable because a lower maximum temperature is preferred for electrical equipment. The change to ARI 640 is evaluated in Section 9.4.1.

9.4.9.3 Conclusion

The NRC staff has reviewed the proposed changes to the VTS system and finds them acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VTS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.10 Diesel Generator Building Heating and Ventilation System (VZS)

9.4.10.1 Summary of Technical Information

In Section 9.4.10.2.1.1, “Normal Heating and Ventilation System,” the applicant proposed to add the following paragraph:

> Electric unit heaters are provided in the diesel generator stairwell and security room to maintain the space at a minimum temperature.

9.4.10.2 Evaluation

The above mentioned DCD change is technically sound. The design change is acceptable.

9.4.10.3 Conclusion

The NRC staff has reviewed the proposed change to the VZS system and finds it acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VZS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.4.11 Health Physics and Hot Machine Shop HVAC System (VHS)

9.4.11.1 Summary of Technical Information

In Section 9.4.11.1.2, “Power Generation Design Basis,” of the DCD the applicant proposed to add the “Security Room,” the “Elevator Machine Room” and the “Stairwell” to the VHS ventilation system coverage rooms/areas with specified temperature ranges.

In Section 9.4.11.2.2, “Component Description,” the applicant proposed to change the humidifier description such that the design and rating of the VHS humidifier will be in accordance with ARI 640.

9.4.11.2 Evaluation

The staff finds this change acceptable because the VHS ventilation system coverage has no safety significance. The change to ARI 640 is evaluated in Section 9.4.1.

9.4.11.3 Conclusion

The NRC staff has reviewed the proposed change to the VHS system and finds it acceptable. On the basis of the evaluation described in NUREG-1793 and this evaluation, the NRC staff concludes that the VHS system is acceptable and that the application for DC meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

9.5.1.1 Summary of Technical Information

Section 9.5.1, "Fire Protection System" of the AP1000 DCD, Revision 15, was approved by the staff in NUREG-1793. In AP1000 DCD, Revision 17, the applicant has proposed to make the following change to Section 9.5.1:

1. In the AP1000 DCD, Revision 17, the applicant proposes to revise Section 9.5.1.8, alleviating the need for the COL applicant to submit additional information and to close out COL Action Items 9.5-5 and 9.5-8.

2. In Table 9.5.1-1, the applicant proposes to add carpeting into the MCR (e.g., for sound abatement or other human factors).

<u>Multiple Spurious Actuations</u>

Section 9A.2.7.1, "Criteria and Assumptions," of Appendix 9A, "Fire Protection Analysis," of the DCD, states the following with respect to the approach to evaluating multiple spurious actuations that result from fire-induced electrical shorts: "spurious actuations or signals resulting from the fire are postulated one-at-a-time (except for high/low pressure interfaces)." However, as noted in RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 1, the "one-at-a-time" assumption for spurious actuations may not adequately address the potential risk attributed to fire as demonstrated by NRC and industry fire tests.

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.5.1.2 Evaluation

In the AP1000 DCD, Revision 17, the applicant proposes to add carpeting into the MCR as allowed by RG 1.189, Regulatory Position 6.1.2. Although the DCD stated that the MCR carpeting issue is addressed in the fire protection analysis, the NRC staff noted that only the additional fire loading has been documented in the fire hazard analysis. Since the introduction of carpeting as proposed in DCD Revision 17 increases the fire duration in Fire Area 1242 AF 01 (MCR Complex) beyond 60 minutes, the current fire protection adequacy evaluation based primarily on the "light hazard" assumption for this fire area may no longer be valid. In addition, since the fire duration in Fire Zone 1242 AF 12401B increases to 75 minutes, which exceeds the 1-hour fire barrier rating between the two fire zones within the MRC Complex, the assumption that a fire is limited to one fire zone within the MCR Complex is no longer valid. Furthermore, while RG 1.189 references American Society for Testing and Materials (ASTM) D2859, "Standard Test Method for Flammability of Finished Textile Floor

Covering Materials," for establishing the acceptable flammability characteristics of the material, the applicant establishes compliance by referencing ASTM E-648, "Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source," and National Fire Protection Association (NFPA) 253, "Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source," flame tests. In reviewing the above standards, the NRC staff cannot establish a direct correlation in the testing methods and acceptance criteria between ASTM D2859 and ASTM E-648 or NFPA 253.

In RAI-SRP9.5.1-SFPB-02, the staff requested that the applicant provide additional information to demonstrate that the alternative fire testing standards meet ASTM D2859 at the minimum. In addition, it was requested that the applicant revise the DCD and provide an evaluation that shows the additional fire loading does not impact the existing fire protection adequacy evaluation for the MCR Complex and it was requested that the applicant justify that the exclusion of a fixed fire suppression system in this fire area is still acceptable.

In a letter dated September 23, 2009, the applicant proposed several DCD revisions to address the information requested in RAI-SRP9.5.1-SFPB-02. The applicant included the testing standard used and revised Table 9.5.1-1 to indicate that the flammability characteristics of the carpeting are acceptable when tested to ASTM D2859. In addition, the applicant has stated that a revised combustible loading/fire severity calculation has been performed using a lower carpet quantity (weight per square foot), based on carpet vendor data, for carpeted areas and further reduced quantities to reflect areas not carpeted. Based on the new calculation, the applicant affirmed that the fire severity in the affected zone/area remains under the 1-hour fire duration, thus the "light hazard" assumption in the affected fire zone/area is maintained. Based on the above, the NRC staff determined that the applicant has adequately addressed the technical concerns in RAI-SRP9.5.1-SFPB-02. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Multiple Spurious Actuations

On February 19, 2009 and again on March 31, 2009, the NRC staff conducted audits of the applicant's fire hazards analysis supplemental report APP-FPS-G1R-002 and discussed the "one-at-a-time" assumption used in the report. The applicant stated that this assumption is not applied in supplemental report APP-FPS-G1R-002 or any other safe-shutdown analyses. The applicant further stated that by design, any unsafe plant conditions created by spurious equipment actuations (regardless of single or multiple spurious actuations) will be detected by the redundant safety instrumentation logics and will be mitigated by operators using the preferred safe-shutdown method or ultimately by the AP1000 passive safe-shutdown system. Since the applicant maintained that the "one-at-a-time" assumption is not used in the fire hazards analysis or any other safe-shutdown analysis, the NRC staff requested in RAI-SRP9.5.I-SFPB-01 that the "one-at-a-time" assumption be replaced with "multiple spurious actuations" assumption in applicable sections of the AP1000 DCD and that supplemental report APP-FPS-G1R-002 be referenced in the DCD. In a letter dated June 9, 2009, the applicant accepted the NRC staff's recommendation and proposed the above changes in the next revision of the DCD. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

9.5.1.2.1 Evaluation of COL Action Items

COL Action Item 9.5-5

In the AP1000 DCD, Revision 17, the applicant proposes to revise Section 9.5.1.8, alleviating the need for the COL applicant to submit additional information and to close out COL Action Item 9.5-5.

COL Action Item 9.5-5:

> The COL applicant will provide an analysis to demonstrate that operator actions that minimize the potential for spurious actuation of the automatic depressurization system (ADS) as a result of a fire can be accomplished within 30 minutes following detection of the fire, as well as the procedure for manual actuation of the fire water containment supply isolation valve to allow fire water to reach the automatic fire system in the containment maintenance floor.

This proposed change is supported by TR-45, "Operator Actions Minimizing Spurious ADS Actuation," APP-GW-GLR-027, Revision 1, dated June 2006, in which the applicant provided an analysis demonstrating that operator manual action can be accomplished within 30 minutes to minimize the potential for spurious automatic depressurization system (ADS) actuation in the event of a fire. The TR also stated that the procedure for the manual actuation of the valve to allow fire water to reach the automatic fire system in the containment maintenance floor has been written.

The ADS consists of four different stages of valves that are designed to open sequentially when actuated and to remain open for the duration of an automatic depressurization event. The sequential valves actuation logic relies on a combination of time delays and level indicators to prevent simultaneous opening of more than one stage at a time, and thus provides for a controlled depressurization of the RCS. Manual actuation switches are also provided in the MCR and on the remote shutdown console (RSC).

The applicant had previously asserted in the AP1000 DCD, and in subsequent responses to the staff's RAIs, that the inherent design of the ADS actuation logic, and the spatial separation of the manual activation switches in the MCR, made fire-induced spurious ADS actuation a highly unlikely event and therefore not a concern. However, the NRC staff has maintained that for certain plant areas, including the MCR, Division A Instrumentation and Control (I&C) Room, Division B I&C Room, Division C I&C Room, Division D I&C Room, Division A Penetration Room, and Division C Penetration Room, fire-induced spurious ADS actuation cannot be precluded due to the potential of multiple hot shorts and smoke-induced integrated circuit failures. Consequently, the applicant introduced post-fire operator actions as prescribed in TR45 to further minimize the probability of spurious ADS actuation in the above plant areas.

The staff initially had a concern regarding the feasibility and reliability of the proposed post-fire operator manual actions prescribed in TR-45 and issued RAI-TR45-006 in a letter dated March 30, 2007. In a letter dated July 13, 2007, the applicant responded to RAI-TR45-006 asserting that although spurious ADS actuation is undesirable, if ADS were to actuate as the result of a fire, the AP1000 plant would still be able to achieve and maintain post-fire safe shutdown. Also, the disabling of the ADS as proposed has no impact on the ability to achieve and maintain post-fire safe shutdown. Since the proposed post-fire operator manual actions are neither required for achieving and maintaining post-fire safe shutdown nor will create an adverse impact on the post-fire safe-shutdown capability, the staff concludes that these actions do not have to meet the feasibility and reliability criteria outlined in RIS 2006-10, “Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions.” Furthermore, these operator manual actions, aimed to minimize ADS actuations, have no adverse impact on the fire protection program.

In addition to providing the analysis supporting the minimization of spurious ADS actuation, TR-45 also provided that the applicant’s document, APP-GW-GJP-305, “AP1000 Fire Emergency Response,” includes steps for manual alignment of valves to allow fire water to reach the automatic fire system in the containment maintenance floor. This procedure, in addition to the above analysis, provided adequate information to satisfy COL Action Item 9.5-5.

<u>COL Action Item 9.5-8</u>

Section 9.5.1.8, “Special Protection Guidelines (Regulatory Position C.8 of BTP cmEB 9.5-1)” of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 17, the applicant proposes to revise Section 9.5.1.8, alleviating the need for the COL applicant to submit additional information and to close out COL Action Item 9.5-8. This proposed change is supported by TR-46, “Fire Resistance Test Data,” APP-GW-GLR-019, Revision 0, dated May 2006, in which the applicant provided the fire resistance test data for the concrete/steel composite building material selected for use in certain firefighting and safe shutdown access/egress routes, in particular the stairwell towers within the Auxiliary Building. COL Action Item 9.5-8:

> The COL applicant will provide 2-hour fire resistance test data in accordance with American Society for Testing and Materials (ASTM) Standard E-119 and NFPA 51 for the composite material selected for stairwell fire barriers.

Regulatory Position C.5.a.6 of Branch Technical Position (BTP) cmEB 9.5.1 specifies that “Stairwells outside primary containment serving as escape routes, access routes, for firefighting, or access routes to areas containing equipment necessary for safe shutdown should be enclosed in masonry or concrete towers with a minimum fire rating of 2 hours and self-closing Class B fire doors.” The AP1000 design, however, deviates from the above guideline by specifying concrete/steel composite material with an equivalent fire resistance rating of 2 hours in lieu of masonry or concrete for the auxiliary building stairway towers. The applicant, however, did not provide test reports to demonstrate that the designed concrete/steel composite configuration would meet the applicable regulation (GDC 3, “Fire Protection”) and the applicable guidance (BTP cmEB 9.5-1) requiring an equivalent level of safety of concrete or masonry. The responsibility to provide test data was deferred to the COL applicant. Consequently, the NRC staff assigned COL Action Item 9.5-8 requiring the COL applicant to provide fire resistance test data in accordance with ASTM E-119, “Standard Test Method of Fire Tests of Building Construction and Materials,” and NFPA 251, “Tests of Fire Endurance of Building Construction and Materials,” for the concrete/steel composite material.

TR-46 supported the close-out of COL Action Item 9.5-8 by providing the fire resistance test data for the concrete/steel composite material. TR-46 confirmed that test results from Factory Mutual Research Corporation’s Report J. I. 1R7Q3 successfully demonstrated at least a 2-hour fire resistance rating for the concrete/steel composite material as required by regulatory Position C.5.a.6 of BTP CMEB 9.5-1. This TR also confirmed that the fire resistance test was performed in accordance with ASTM E-119 and NFPA 251 as required per Section 3.1.6 of Generic Letter (GL) 86-10, “Implementation of Fire Protection Requirements.”

Based on the above, the staff concluded that the applicant has provided adequate information to demonstrate the adequacy of the concrete/steel composite material to provide a fire resistance equivalent to that of a 2-hour rated concrete or masonry barrier as specified in Regulatory Position C.5.a.6 of BTP CMEB 9.5-1. Accordingly, COL Action Item 9.5-8 can be closed.

9.5.1.3 Conclusion

The NRC staff finds that the applicant has adequately addressed the staff's concern regarding the additional combustible loading/fire severity in the MCR Complex. The NRC staff also finds that the applicant's proposed DCD revision to replace the fire-induced "one-at-a-time" assumption with the "multiple spurious actuations" assumption is acceptable. The staff also concludes that the applicant has provided adequate information to close COL Action Item 9.5-5 and COL Action Item 9.5-8.

Based on the evaluation above, the staff concludes that the proposed changes are acceptable because the AP1000 design continues to meet the applicable requirements and guidance in GDC 3, BTP CMEB 9.5-1 and 10 CFR 50.48. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii), because they contribute to increased standardization of the certification information.

9.5.2 Communications Systems

9.5.2.1 Summary of Technical Information

The staff has reviewed the amendments to Section 9.5.2 of the AP1000 DCD, in accordance with the acceptance criteria in NUREG-0800 Section 9.5.2, "Communications System."

In Section 9.5.2.2.3, "Private Automatic Branch Exchange (PABX) System," the applicant has proposed to change the PABX interface to communications system requirements with the following modifications:

The applicant states that the hotlines to specified locations (for example, dedicated communication lines with load dispatcher to support and coordinate the system gird) are as described in Section 9.5.2.5, "Combined License Information." The hotline circuits are dedicated channels that provide direct communication between the MCR and the headquarters, or other facilities, are as specified in Section 9.5.2.5.

Direct extensions from the PABX locations exterior to the plant is as dictated by Section 9.5.2.5.

In addition, the applicant has proposed to modify the requirements for commercial telephone lines that are provided by the local area telephone company. Specifically, the applicant specifies that the number of lines will be defined as required in Section 9.5.2.5.

9.5.2.2 Evaluation

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

In RAI-SRP9.5.2-ICE-01, the staff requested additional information regarding the transfer of requirements from Section 9.5.2.2.3 to Section 9.5.2.5 for the COL applicant to address PABX interfaces to hotlines and direct extensions from the PABX locations exterior to the plant. In the July 31, 2008, response to the staff's request for additional information, the applicant provided justification for these modifications. The applicant states that the changes made in Section 9.5.2.2.3 were administrative as described in TR-130, "Editorial Format Changes Related to Combined License Applicant and COL Information Items," APP-GW-GLR-130, Revision 0 of June 2007. TR-130 states:

"Up through Revision 15 of the DCD, there were many instances where the text states, "...the Combined License applicant will...." or words exist stating something similar. The applicant has taken the position that in the "Combined License Information" sections in Tier 2 of the DCD, these words are appropriate. Having the words appear in other sections of the DCD, however, leads to confusion, especially when the COL applicant is attempting to incorporate by reference the DCD section (or subsection) into their individual COL application. As a result, the applicant has reviewed the DCD and has removed these types of phrases from Tier 2… "In no case is it the intent of Westinghouse to change the commitment that exists in DCD, as the changes are intended to be editorial."

9.5.2.3 Conclusion

The applicant states that the requirements in Section 9.5.2.2.3 have not been modified or removed nor have they been transferred to Section 9.5.2.5. As delineated in Section 9.5.2.5, the COL applicant will describe how it meets the requirements specified in Section 9.5.2. The staff finds this response acceptable. The staff verified that the modifications made in Section 9.5.2.2.3 are administrative and do not affect the requirements within the COL action items described in Section 9.5.2.5. The staff concludes that the changes made in Revision 19 of the AP1000 DCD do not impact the staff's conclusions within Section 9.5.2, "Communication Systems" of NUREG-1793 regarding the AP1000 communication system's compliance. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii); on the basis that they contribute to increased standardization of the certification information and are, therefore, acceptable.

9.5.4 Diesel Generator Auxiliary Support Systems

9.5.4.1 Summary of Technical Information

Section 9.5.4, "Standby Diesel Fuel Oil System" of the AP1000 DCD, Revision 15, was approved by the staff in NUREG-1793. In AP1000 DCD, Revision 17, the applicant has proposed to make the following changes to Section 9.5.4 of the certified design:

1. The applicant has proposed to delete the function of supplying diesel fuel oil to the auxiliary boiler. In the previously approved AP1000 DCD, Revision 15, the standby diesel fuel oil system supplied fuel oil to the auxiliary boiler.

2. The applicant has proposed to complete COL Information Item 9.5-12 of Table 1.8-2, which is addressed in Section 9.5.4.7 of the AP1000 DCD Revision 15 and states:

    > Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

3. The applicant has proposed to partially complete COL Item 9.5-13 of Table 1.8-2, which is addressed in Section 9.5.4.7 of the AP1000 DCD Revision 15 and states:

    Combined License applicants referencing the AP1000 certified design will address site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

9.5.4.2 Evaluation

The staff reviewed all changes to the Standby Diesel Fuel Oil System (SDFOS) in the AP1000 DCD Revision 17 in accordance with the guidance in NUREG-0800 Section 9.5.4, "Emergency Diesel Engine Fuel Oil Storage and Transfer System." The regulatory basis for Section 9.5.4 of the AP1000 DCD is documented in NUREG-1793. The following evaluation discusses the results of the staff's review.

The specific criterion that applies to the changes referred to above is 10 CFR 52.63(a)(1)(vii), which concerns contribution to increased standardization of the certification information.

9.5.4.2.1 Delete the Function of Supplying Diesel Fuel Oil to the Auxiliary Boiler

In DCD Revision 17, the applicant changed Tier 1 Section 2.3.3, ITAAC Table 2.3.3-2; and Tier 2 pages 9.5-24 through 9.5-28, Table 9.5.4-1, Table 9.5.4-2, and Figure 9.5.4-1. The applicant deleted the function of supplying diesel fuel oil to the auxiliary boiler. In TR-114, "AP1000 Auxiliary Boiler Sizing and Design," APP-GW-GLN-114, Revision 0 of June 2007, the applicant states that utilities have reported operational problems due to fouling of the fuel in diesel boilers that sit idle for extended periods of time. Some have changed their auxiliary boilers to electric boilers. For AP1000, the applicant has proposed a design change from a diesel fired auxiliary boiler to an electric auxiliary steam boiler. The fuel oil pumps and fuel oil piping associated with the auxiliary boiler have been removed from the DCD. The staff finds this change does not affect the function of the SDFOS to supply diesel fuel oil to the standby DGs. The staff also finds that the statements regarding the auxiliary boiler fuel oil supply in Section 9.5.9 of NUREG-1793 are no longer applicable. Since the changes to the SDFOS, which are the removal of the portion that would supply the auxiliary boiler, do not affect the function to supply fuel oil to the standby DGs, the staff finds that the conclusions of NUREG-1793 regarding the acceptability of the SDFOS remain valid.

9.5.4.2.2 Resolution of COL Item 9.5-12

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 9.5-12 which addresses cathodic protection of diesel fuel oil storage tanks. COL Information Item 9.5-12 in the DCD is also discussed in NUREG-1793 as COL Action Item 9.5.9-1. The applicant submitted TR-120, "Cathodic Protection for Metal Tanks in Contact With the Ground," APP-GW-GLR-120, of 25 May, 2007, for staff review to close out COL Information Item 9.5-12. The proposed change will eliminate the need for COL applicants to address cathodic protection of diesel fuel oil storage tanks as stated in COL Information Item 9.5-12.

In Revision 15, Section 9.5.4.7 to the AP1000 DCD, COL Information Item 9.5-12 states:

> Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

This COL item refers to National Association of Corrosion Engineers (NACE International) Standard Recommended Practice RP0169, “Control of External Corrosion on Underground or Submerged Metallic Piping Systems,” which is referenced in RG 1.137, “Fuel-Oil Systems for Standby Diesel Generators.” Since the diesel fuel tanks proposed for the AP1000 are on grade rather than buried, TR-120 proposes to address the COL item based on an alternative to NACE International RP0169.

In Revision 17 of the AP1000 DCD, the applicant proposed to resolve COL Information Item 9.5-12 by addressing the cathodic protection of diesel fuel oil storage tanks in TR-120, Section 9.5.4.7 of the DCD was revised to add Section 9.5.4.7.1, which states:

> The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-120 (Reference 24), and the applicable changes are incorporated into the DCD. No additional work is required.
>
> The following words represent the original COL Information item commitment, which has been addressed as discussed above:
>
>> Combined License applicants referencing the AP1000 certified design will address the site-specific need for cathodic protection in accordance with NACE Standard RP-01-69 for external metal surfaces of metal tanks in contact with the ground.

According to Section 9.5.4.2.2.1 of the DCD, the fuel oil storage tanks are located on grade and will be erected on a continuous concrete slab contained within a concrete dike. The tanks will have no direct contact with the soil, and the design is intended to minimize the intrusion of groundwater and rainwater into the interface between the tank bottom and concrete foundation.

In its report, the applicant explained that because NACE International RP0169 applies to underground tanks and the AP1000 diesel tanks are above ground, this COL item is being addressed according to guidance in American Petroleum Institute (API) Recommended Design Practice 651 (API 651), “Cathodic Protection of Aboveground Storage Tanks.” The applicant quoted a portion of Section 5.3.3 in API 651 that indicates: (1) a properly designed concrete tank pad may be effective in eliminating external corrosion from the soil and the need for cathodic protection (CP); and (2) moisture may still collect between the tank bottom and pad, and CP is generally not an effective corrosion control method under these conditions. The corrosion protection proposed for these tanks is, therefore, based on a combination of keeping the tank bottom dry and coating it with an appropriate epoxy-urethane paint system.

Because cathodic protection of a structure relies on the flow of electrical charge (i.e., current) from external electrodes (anodes) to the structure (cathode), successful application of CP requires an electrolyte path between the anodes and the structure. (See, for example, “Cathodic Protection,” in Metals Handbook, Ninth Edition, Volume 13, ASM International, 1987.) For external protection, CP is normally used in a soil or aqueous environment in conjunction with an external coating on the structure; hence, current is needed only at coating defects that expose the underlying metal. For the AP1000 diesel tanks, the staff concludes CP would not be useful under the design conditions because the tank bottoms are expected to be dry. Any anticipated moisture accumulation causing partial or intermittent wetness would not be expected to cause significant corrosion of tanks utilizing a well designed and maintained coating system.

The staff notes that another NACE International Standard Recommended Practice, RP0193, "External Cathodic Protection of On-Grade Carbon Steel Storage Tank Bottoms," provides guidance similar to API 651:

> 5.7 On-grade tanks that are set on solid concrete or asphalt pad foundations generally require specialized measures for corrosion protection, because cathodic protection may be ineffective. In this circumstance, the external surface of the tank bottom should be coated. In all cases, steps should be taken to ensure that water does not migrate between the tank bottom and the pad.

The industry recommended practices cited by the applicant and by the staff indicate cathodic protection is not required for the on-grade diesel fuel storage tanks proposed for the AP1000, and the proposed resolution of COL Information Item 9.5-12 is, therefore, technically sound. Therefore, the staff found that the DCD changes, as proposed by the applicant in TR-120, are acceptable, and COL Information Item 9.5-12 is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes incorporated into Revision 17 contribute to the increased standardization of the certification information in the AP1000 DCD.

The staff notes that the standby DGs and their support systems (e.g., fuel storage tanks) have no safety-related functions in the AP1000 passive plant design and, therefore, no nuclear-safety design basis. Hence, storage tank design features discussed above are not safety significant. NUREG-1793, which was written based on Revision 14 of the AP1000 DCD, stated this conclusion as follows:

> Based on its review, the staff determined that the DG and auxiliary boiler fuel oil system is a non-safety-related system and serves no safety-related function. Its failure does not lead to the failure of any safety systems. The staff, therefore, concludes that the requirements of GDC 2, 4, 5, and 17, and the guidance of SRP Section 9.5.4, do not apply.

9.5.4.2.3 COL Item 9.5-13

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve the portion of COL Information Item 9.5-13, that addresses the need to reduce heat input to the fuel oil storage tanks from the sun. COL Information Item 9.5-13 in the DCD is also discussed in NUREG-1793 as COL Action Item 9.5.9-2, which is associated with ensuring the quality of diesel fuel oil. The applicant submitted TR-120 for staff review to close out COL Information Item 9.5-13. The proposed change will eliminate the need for COL applicants to address the site-specific requirement for reducing heat input to the fuel oil storage tanks from the sun. With respect to the other two parts of COL Information Item 9.5-13, the proposal is clear that addressing the fuel specifications grade, properties, and sampling/testing program remains the responsibility of the COL applicant.

In Revision 15, Section 9.5.4.7 to the AP1000 DCD, COL Information Item 9.5-13 states:

> Combined License applicants referencing the AP1000 certified design will address site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

COL Information Item 9.5-13 addresses three separate issues: 1) reducing heat input to the tank from the sun; 2) specifying the proper fuel grade and properties based on the engine manufacturer recommendations; and 3) protecting against fuel degradation with a fuel sampling and testing program. The applicant proposes to partially resolve this COL item by closing out that portion of the COL item related to reducing heat input to the tank from the sun. COL Information Item 9.5-13 will remain open and COL applicants are required to specify the proper fuel grade and properties based on the engine manufacturer's recommendations, and provide measures to protect against fuel degradation with a fuel sampling and testing program.

In Revision 17 of the AP1000 DCD, the applicant proposed to partially resolve COL Information Item 9.5-13 by addressing heat input to the tank from the sun in TR-120. Section 9.5.4.7 and Table 1.8-2 of the DCD was revised to add Section 9.5.4.7.2, which states:

> 9.5.4.7.2 The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-120, (Reference 24), and the applicable changes are incorporated into the DCD. No additional work is required to address the information requested in this subsection as delineated in the following paragraph:
>
>> The epoxy-urethane paint color selected for the exterior of the standby diesel fuel oil storage tanks shall be white to minimize radiant sunlight heat transmission to the tank oil stored fuel volume.
>
> The following activities are to be addressed by the Combined License applicant:
>
>> Address the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations and the measures to protect against fuel degradation by a program of fuel sampling and testing.
>
> The following words represent the original Combined License Information item commitment, which has been addressed as discussed above:
>
>> Combined License applicants referencing the AP1000 certified design will address site-specific factors in the fuel oil storage tank installation specification to reduce the effects of sun heat input into the stored fuel, the diesel fuel specifications grade and the fuel properties consistent with manufacturers' recommendations, and will address measures to protect against fuel degradation by a program of fuel sampling and testing.

Heat input to the tank is addressed by stating that the color of the epoxy-urethane exterior coating will be white to minimize heat input to the tank contents. This approach is technically sound. For example, it is consistent with ASTM D-975, “Standard Specification for Diesel Fuel Oils,” – 07b, Section X3.7, which states reflective paint color should be used to prevent thermal degradation of the fuel oil.

With respect to the other two parts of COL Information Item 9.5-13, the proposal is clear that addressing the fuel specifications grade, properties, and sampling/testing program remains the responsibility of the COL applicant.

Based on the above, the staff finds that the portion of COL Information Item 9.5-13 that addresses reducing heat input to the tank from the sun is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design.

9.5.4.3 Conclusion

Based on the above evaluation, the staff finds the applicant’s proposed design change from a diesel fired auxiliary boiler to an electric auxiliary steam boiler does not affect the function of the SDFOS to supply diesel fuel oil to the standby DGs and is, therefore, acceptable. The staff also finds that the applicant’s proposed resolution to COL Information Item 9.5-12 is consistent with applicable industry guidance and is, therefore, acceptable. The staff also finds that the applicant’s proposed partial resolution to COL Information Item 9.5-13 for reducing heat input to the tank from the sun is consistent with applicable industry guidance and is acceptable. The proposed changes meet the criterion of 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to increased standardization of the certification information.

10. STEAM AND POWER CONVERSION SYSTEM

10.1 Introduction

Westinghouse Electric Company LLC (Westinghouse or the applicant) proposed to the U.S. Nuclear Regulatory Commission (NRC) an alternative design for its steam and power conversion (SPC) system in its technical report (TR)-86, “APP-GW-GLN-018, Revision 0, “Alternate Steam & Power Conversion Design,” dated February 8, 2007. On June 25, 2007, the applicant submitted Revision 1 to TR-86 and proposed a new standard design with a single turbine-generator and steam-cycle unit for all AP1000 plants. In Revision 17 to the AP1000 design control document (DCD), Tier 2, Section 10.2, “Turbine Generator,” the applicant incorporated all the changes associated with the new standard design.

The most significant differences between the designs that the NRC staff evaluated in NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” issued September 2004, and the proposed new design of the main turbine-generator set are as follows:

(1) replacement of the mechanical overspeed protection with a diverse electrical overspeed trip;

(2) triplicate channels for turbine speed indication and turbine trip signal versus two redundant channels;

(3) three modes of turbine overspeed protection versus two modes;

(4) hydraulic trip manifold, which enables online testing;

(5) addition of 7th stage feedwater heaters;

(6) decrease in diameter from 137 centimeters (cm) (54 inch (in)) to 132 cm (52 in) for the low-pressure turbine last stage blade (LSB);

(7) addition of moisture extraction blades (MEB); and

(8) static excitation provided by solid-state thyristors versus brushless excitation.

In the AP1000 DCD, Tier 1, Section 2.4.2, Table 2.4.2-2, the applicant proposed adding an “emergency electrical overspeed trip device” as a component located in the turbine building.

In addition, AP1000 DCD, Section 10.2, includes the following proposed changes:

- In DCD Tier 2, Section 10.2.6, “Combined License Information on Turbine Maintenance and Inspection,” the applicant proposed revising the implementation timing for combined license (COL) Information Item 10.2.6. This was identified as COL Action Item 10.5-2 in Section 10.5 of Appendix F, “Combined License Action Items,” to NUREG-1793.

- In DCD Tier 2, Section 10.2.3.6, “Maintenance and Inspection Program Plan,” the applicant proposed revising the turbine valve testing intervals from quarterly to semiannually.

- In DCD Tier 2, Section 10.2, the applicant proposed a number of changes related to the layout and general arrangement in the turbine building as a result of replacing the original design with the alternative design, which includes an additional stage of feedwater heaters.

- In DCD Tier 2, Section 10.2.2.4, “Digital Hydraulic System Description”; Section 10.2.2.5, “Overspeed Protection”; and Table 10.2-2, “Turbine Overspeed Protection”; the applicant proposed revisions to reflect the replacement of the Toshiba Turbine Control System with an Ovation® Turbine Control System.

- In DCD Tier 2, Figure 10.2-1, “Turbine Generator Outline Drawing,” the applicant revised the drawing to provide consistency with the Toshiba Turbine-Generator Steam Cycle Design.

All other changes to Section 10.2 were determined by the staff to be editorial in nature and do not affect the conclusion in NUREG-1793 and are, therefore, acceptable.

10.2 Turbine Generator

The staff reviewed all changes to the turbine generator, in accordance with Section 10.2, “Turbine Generator,” of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants.” The new design of the turbine-generator system was evaluated against the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic licensing of production and utilization facilities.” Specifically, the design must meet the requirements of General Design Criterion (GDC) 4, “Environmental and Dynamic Effects Design Bases,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, as they relate to protecting the structures, systems, and components (SSCs) that are important to safety from the effects of turbine missiles. GDC 4 provides guidance for the turbine overspeed protection system (with suitable redundancy and diversity) to minimize the generation of turbine missiles. NUREG-0800 Section 10.2.II, “Acceptance Criteria,” describes the specific criteria needed to meet the requirements of GDC 4. Also, the staff used NUREG-0800 Section 10.2.III, “Review Procedures,” to determine whether the preliminary design in the DCD meets the acceptance criteria in NUREG-0800 Section 10.2.II.

The staff focused its review on changes 1 through 4 identified above, COL Action Item 10.5-2, turbine valve testing, and the general layout arrangements, since these changes affect the protection of the turbine from overspeed conditions that could lead to the generation of turbine missiles. These items relate directly to the specific criteria described in Items 1 A and 1 C of NUREG-0800 Section 10.2.II that are necessary to meet the requirements of GDC 4, as they relate to the turbine overspeed protection and periodic testing of components while the unit is operating at rated load. Also, the staff reviewed changes 5 through 8 identified above and determined that these changes do not affect the safety conclusions in NUREG-1793.

For the purpose of this evaluation, the NRC refers to the design of the turbine-generator system provided in the AP1000 DCD, Revision 15, and evaluated in Section 10.2 of NUREG-1793, as the "original" design. The staff refers to the alternative Toshiba design in the proposed changes to the AP1000 DCD as the "new" design.

The safety design bases and power generation design bases for the turbine generator are the same for both the original and new designs. The turbine generator of the new design is designated as Model TC6F, with a diameter of 132 cm (52 in) for the low-pressure turbine LSB unit. The low-pressure turbine LSB in the original design had a diameter of 137 cm (54 in). DCD Tier 2, Table 10.2-1, identifies the design parameters of the new design. DCD Tier 2, Figure 10.3.2-2, provides the piping and instrumentation diagram containing the stop, control, intercept, and reheat valves. Also, the new design consists of a double-flow, high-pressure turbine; three double-flow, low-pressure turbines; and two external moisture separator/reheaters. The difference between the two designs is that the new design has two reheating stages, while the original design had only one.

The single direct-driven generator is cooled by hydrogen gas and de-ionized water (water-cooled in the original design). Other related system components include a turbine-generator-bearing lubrication system, a digital electrohydraulic control (D-EHC) system with supervisory instrumentation, a turbine steam-sealing system, overspeed protective devices, turning gear, a stator cooling-water system, a generator hydrogen and seal oil system, a generator carbon-dioxide system, a rectifier section, an excitation transformer, and a voltage regulator. The D-EHC system, the overspeed protective devices, and the excitation system in the new design differ from those in the original.

The turbine-generation foundation in both designs is a spring-mounted support system. The springs dynamically isolate the turbine-generator deck from the remainder of the structure in the range of operating frequencies.

The turbine-generator cycle is basically the same for both designs. Steam from each of the two steam generators enters the high-pressure turbine through stop valves and control valves (CVs). After expanding through the high-pressure turbine, exhaust steam flows through two external moisture separator/reheater vessels. The reheated steam flows through separate reheat stop and intercept valves in each of six reheat steamlines leading to the inlets of the three low-pressure turbines. Turbine steam extraction connections are provided for seven stages of feedwater heating (six stages in the original design). Moisture separation in the new design differs from the original; however, the difference does not affect the evaluation and the staff's conclusion in NUREG-1793.

10.2.1 Overspeed Protection

A D-EHC system and an emergency trip system provided overspeed protection in the original design. The D-EHC system opened a drainpath for the hydraulic fluid in the overspeed protection control header, if the turbine exceeded 103 percent of rated speed. As a result of the loss of fluid pressure in the header, the control and intercept valves closed.

The emergency trip system tripped the turbine if speeds exceeded 110 percent of the rated speed. The emergency trip system has an emergency trip control block, trip solenoid valves, a mechanical overspeed device, three test trip blocks with pressure sensors and test solenoid valves, rotor position pickups, speed sensors, and a test panel. The mechanical overspeed trip device consisted of a spring-loaded trip weight mounted in the rotor extension shaft.

In the new design, the two systems that provide overspeed protection, the D-EHC control system and the electrical overspeed trip system, differ from those in the original.

The new D-EHC has three modes of operation to protect the turbine generator against overspeed. The first is the intercept-valves control function, which initiates closure of the intercept valves when the error between the demand position signal and the actual position signal of the intercept valve exceeds the setpoint. The second mode is load unbalance, which operates at greater than or equal to 30-percent load rejection. It causes all control and intercept valves to close quickly. The third mode is the emergency overspeed trip. All CVs and intercept valves are fully closed quickly by the actuation of each fast-acting solenoid valve of the electrical overspeed trip system.

The electrical overspeed trip system consists of redundant processors, three speed channel circuits, and trip relays. An independent electrical overspeed system replaces the mechanical overspeed device of the original design. A trip of the system opens a drainpath for the hydraulic fluid in the emergency trip supply. A loss of fluid pressure in the trip header causes the main stop and reheat valves to close. Also, a relay trip valve in the connection to the emergency trip supply opens to drop the pressure in the relay emergency trip supply and cause the control and intercept valves to close.

The specific criteria that apply to the overspeed protection related changes include 10 CFR 52.63(a)(1)(vi), “Finality of standard design certifications,” in that the changes substantially increase overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

The following Sections 10.2.2 and 10.2.2.1 provide further details on the D-EHC and the electrical overspeed trip systems, respectively.

10.2.2 Digital Electrohydraulic Control System

AP1000 DCD, Section 10.2.2.4, “Digital Electrohydraulic System Description,” states that the turbine generator is equipped with a D-EHC system that combines the capabilities of redundant processors and high-pressure hydraulics to regulate steamflow through the turbine. The system provides the functions of speed control, load control, and automatic turbine control. A hydraulic system, independent of the bearing lubrication system, provides valve opening actuation in the D-EHC system. Upon reduction or relief of fluid pressure, springs provide valve-closing actuation. The system is designed so that loss of fluid pressure, for any reason, leads to valve closing and, consequently, a turbine trip. Steam valves are provided in an in-line configuration. The overspeed trip system trips the stop valves; the CVs are modulated by the control system and are actuated by the trip system.

The speed control function of the D-EHC master controller provides speed control, acceleration, and overspeed protection. The speed error signal is derived by comparing the desired setpoint with the actual speed of the turbine. A failure of one speed input generates an alarm. The failure of two or more speed inputs generates an alarm and trips the turbine.

The speed control function exists in triplicate channels. The master controller speed function also contains 110-percent and 111-percent overspeed trips. The 110-percent trip signal is sent to a fast-acting solenoid valve in the hydraulic trip manifold that actuates closure of the stop, control, intercept, and reheat valves. An independent emergency electrical trip is available at 111-percent turbine speed, as a backup to the 110-percent electrical overspeed trip.

The load control function of the D-EHC develops signals that are used to regulate unit load. Signal inputs are based on a proper combination of the speed error and actual load setpoints (turbine megawatt) to generate flow demand on the CVs.

The automatic turbine control provides start up and loading of the turbine generator. The automatic turbine control programs monitor the applicable limits and precautions. When the operator selects the automatic turbine control, the programs both monitor and control the turbine. The automatic turbine control is capable of performing the following activities:

- changing speed
- changing acceleration
- generating speed holds
- changing load rates
- generating load holds

The staff's evaluation finds that the D-EHC system combines the capabilities of redundant processors and high-pressure hydraulics to regulate steamflow through the turbine. The control system provides the functions of speed control, load control, and the automatic turbine control, which may be used either for control or for supervisory purposes, at the option of the plant operator.

The D-EHC system employs three electric speed inputs with signals that are processed in redundant processors. A hydraulic system that is independent of the bearing lubrication system provides valve-opening actuation. Upon reduction or relief of fluid pressure, springs and steam forces provide valve-closing actuation. The system is designed so that a loss of fluid pressure, for any reason, leads to valve closing and consequent turbine trip.

The staff discusses the acceptability of the redundancy and diversity functions of the new D-EHC system design below.

The overspeed trips for the turbine consist of a 110-percent electrical trip and a 111-percent independent emergency electrical trip. In DCD Tier 2, Section 10.2.2.4.1, the applicant stated that the independent emergency electrical trip was available at 111-percent turbine speed, as a backup to the 110-percent electrical overspeed trip.

The electrical overspeed trip system includes an online testable hydraulic manifold, speed sensors, a trip relay, independent power supplies, and a test panel. The emergency trip supply pressure is established when the master trip solenoid valves are closed. The valves are arranged in two channels for testing purposes. Both valves in a channel will open to trip the channel. Both channels must trip before the emergency trip supply pressure collapses to close the turbine steam inlet valves. Each tripping function of the electrical emergency trip system can be individually tested either by the operator or from the test panel, without tripping the turbine, by separately testing each channel of the appropriate trip function. The solenoid valves may be individually tested.

The 110-percent electrical overspeed trip system has triplicated (i.e., redundant) speed sensors, separate from the 111-percent emergency electrical overspeed trip speed sensors, that provide backup overspeed protection, and use the master trip solenoid valves in the master trip device to drain the emergency trip hydraulic supply. The hydraulic fluid in the trip and overspeed protection control headers is independent of the bearing lubrication system to minimize the potential for contamination of the fluid.

Separate instruments sense low bearing oil pressure, low electrohydraulic fluid pressure, and high condenser back pressure. Each assembly consists of triplicate pressure transmitters with instrument valves. Each assembly is arranged into three channels. If two of the three signals (pressure or vacuum) reach a trip setpoint, then the pressure sensors cause the master trip device to operate. A test device can check the trip function by simulating pressure to activate the trip outputs from the modules.

The mechanical overspeed device, evaluated in NUREG-1793, has been replaced with an independent, diverse electrical overspeed trip system. The electrical overspeed trip system consists of redundant processors, three speed channel circuits, and trip relays, as well as other protective functions, such as trip anticipations and power load unbalance detection. The overspeed controllers execute offline and online testing, both of which are conducted from the control room and require no technicians at the turbine. Offline testing is performed during startup and trips the turbine, based on an internal setpoint rather than actual turbine speed. Online testing is automatically executed through the internal injection of a ramping signal into all three independent speed channels at once. This test verifies the proper operation of the software, the hardware, and the components of the hydraulic trip manifold. Loss of one signal will neither cause nor prevent a trip; however, failure of two of the three channels will initiate a turbine trip.

Regarding redundancy, Westinghouse stated that its design achieves additional redundancy by using three speed channel circuits in the master controller of the D-EHC turbine control system. The master controller is the primary controller for starting, synchronizing, and megawatt-loading the turbine/generator. This emergency overspeed trip system residing in the master controller is redundant, independent, and diverse to the electrical overspeed trip system, as discussed above. Each speed channel also has its own independent tripping relays. Loss of one signal will neither cause nor prevent a trip. Further, Section 3.0, "Technical Background," of TR-86, Revision 1, states: "Both the electrical overspeed trip system and the emergency overspeed trip systems in the master controller meet the single-failure criterion and are testable when the turbine is in operation." However, the staff's review of the DCD markup could not find any commitment for either the electrical overspeed trip system or the emergency overspeed trip system in the master controller to meet the single-failure criterion. Item 1.A of NUREG-0800 Section 10.2 states that the overspeed protection system should meet the single-failure criterion and should be testable when the turbine is in operation. Therefore, in request for additional information (RAI)-SRP10.2-SBPA-01, the staff asked the applicant to provide additional information and justification for its claim that its design meets the NUREG-0800 guidance for the single-failure criterion.

In its response to RAI-SRP10.2-SBPA-01, in a letter dated June 20, 2008, the applicant stated that the detailed design of the emergency overspeed trip system for AP1000 is still being completed and that the overspeed protection system design will meet the NUREG-0800 guidance for the single-failure criterion. The applicant is to provide a date for the completed design. The staff had identified this as Open Item OI-SRP10.2-SBPA-01.

The applicant addressed the single failure criteria in its response to RAI-SRP10.2-SBPA-02, dated June 12, 2009. The response included a mark-up of DCD Tier 2, Section 10.2.2.5.3

stating that the overspeed protection system will function for abnormal conditions, including a single failure of any component or subsystem. The staff finds the DCD mark-up meets the NUREG-0800 guidance and is acceptable. Therefore, Open Item OI-SRP 10.2-SBPA-01 is considered closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

With respect to the diversity of overspeed protection, in an earlier request, in Item 3 of RAI-TR86-SBPB-01, the staff asked the applicant to compare the reliability of the proposed turbine overspeed protection capability to the reliability that is afforded by the diverse capability for existing plants. In its letter response dated July 27, 2007, the applicant stated: "Another degree of diversity is provided by the software based trip that takes the speed reading from the input/output (I/O) modules and applies control builder logic to determine the trip function which is then output via separate relay modules." The applicant's response did not specify whether this applies to the primary overspeed trip of 110 percent, or the emergency backup overspeed trip of 111 percent, or both. The staff could not find any further details on the software configuration for the overspeed trip system, either in the changes to the DCD identified in TR-86 or in the responses to other RAIs. The staff's concern is that, if both the 110-percent and 111-percent overspeed trips use the same software, a common-cause failure (CCF) could render both systems inoperable. Also, Item 2.A of NUREG-0800 Section III states that the defense-in-depth provided by the turbine-generator protection system to preclude excessive overspeeds should include diverse protection means. Therefore, with respect to diverse protection against CCF in its design, the staff asked the applicant, in RAI-SRP10.2-SBPA-02, to provide further details to show that it meets the NUREG-0800 guidance, as described above.

In response to RAI-SRP10.2-SBPA-02, the applicant stated that the original design approach was to use Ovation® speed detector module firmware for both trips in parallel with an Ovation® controller software-based logic that provides a level of redundancy and diversity. Now, the applicant has committed to implementing the two overspeed trips using diverse (hardware and software/firmware) electronic means (i.e., one of the trips will not be implemented using the Ovation® speed detector module), so that the 110-percent and 111-percent trips are not susceptible to a common-cause software failure that would render them both inoperable. The staff finds that this commitment meets the guidance described in Item 2.A of NUREG-0800 Section 10.2.III, which states: "The design of the in-depth defense provided by the turbine-generator protection system to preclude excessive overspeeds should include diverse protection means." The staff finds this acceptable; however, the staff's position is that the applicant should update the Tier 1 and Tier 2 sections of the DCD, with inspections, tests, analyses, and acceptance criteria (ITAAC), to confirm that the design acceptance criteria requiring diverse hardware, firmware, and software between the two overspeed trips are met. The staff had identified this as Open Item OI-SRP10.2-SBPA-02a.

Revision 3 of the applicant's response to RAI-SRP10.2-SBPA-02a, dated December 28, 2009, addresses Open Item OI-SRP10.2-SBPA-02a. The applicant provided a mark-up of DCD Tier 1 Table 2.4.2-1 (ITAAC) where Design Commitment 3 has been added, which addresses the applicant's commitment to provide adequate diversity between the two electrical overspeed trips. Design Commitment 3 states: "The trip signals from the two turbine electrical overspeed protection trip systems within the PLS are isolated from, and independent of, each other." For the inspections, tests, and analyses of this ITAAC, the applicant identified three subparts, which include: i) the system design review; ii) testing of the as-built system; and iii) an inspection to be performed for the existence of a report verifying that the two turbine electrical overspeed protection systems have diverse hardware and software/firmware. Further, the applicant described an associated acceptance criteria for each of these inspections, tests, and analyses, where the Acceptance Criteria 3.iii stating: "A report exists and concludes that the two electrical overspeed protection systems within the PLS have diverse hardware and software/firmware." The staff finds this ITAAC acceptable since it ensures that the two electrical overspeed protection systems consist of diverse hardware and software/firmware.

Additionally, in the same December 28, 2009 response, the applicant provided a mark-up of DCD Tier 2, Section 10.2.2.5.3, where it states that the 110 percent and 111 percent trip systems have diverse hardware and software/firmware to eliminate CCFs from rendering the trip functions inoperable. Also in the response, the applicant added Figure 10.2-2, "Emergency Trip System Functional Diagram," to the end of DCD Tier 2, Section 10.2, which shows schematically the diverse hardware and software/firmware of the overspeed trip systems. The staff finds the DCD Tier 1 and Tier 2 mark-ups and added Tier 2 Figure 10.2-2 acceptable as they meet the guidance in Item 2.A of NUREG-0800 Section 10.2.III. Therefore, Open Item OI-SRP 10.2-SBPA-02a is considered closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Also, in NUREG-0800 Section 10.2.III, Item 2.D, the guidance states: "An independent and redundant backup electrical overspeed trip circuit senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 112 percent of rated speed." However, the the applicant response to the above RAI does not state whether the electrical backup senses the turbine speed by magnetic pickup. The staff had identified this as Open Item OI-SRP10.2-SBPA-02b.

In its response to RAI-SRP10.2-SBPA-02, dated June 12, 2009, the applicant provided a mark-up of DCD Tier 2, Section 10.2.2.5.3 where it is stated that an independent and redundant backup electrical overspeed trip circuit senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 111 percent of rated speed. The staff finds the DCD mark-up acceptable as it meets the guidance in NUREG-0800 Section 10.2.III-2c. Therefore, Open Item OI-SRP 10.2-SBPA-02b is considered closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In RAI-SRP10.2-SBPA-04, the staff asked the applicant to provide a timeframe for completion of the system design and submittal of an analysis of failure modes and effects; specifically, common-mode failures of the software for the 110-percent and 111-percent overspeed protection features. In response to RAI-SRP10.2-SBPA-04, the applicant stated that, based on its response to RAI-SRP10.2-SBPA-02 on the diversity of the two turbine overspeed trips provided for the 110-percent and the 111-percent speed points, an analysis would not be necessary; the staff finds this acceptable.

Further, NUREG-0800 Section 10.2.III, Item 2.B, states: "For normal speed-load control, the speed governor of the electrohydraulic control system fully cuts off steam at approximately 103 percent of rated turbine speed by closing the control and intercept valves." The original design included this provision, as shown in Table 10.2-2 in Revision 15 of the AP1000 DCD. However, for the new design, the applicant eliminated the 103-percent trip without providing any justification in the revised DCD or in TR-86, Revision 1. Therefore, in RAI-SRP10.2-SBPA-03, the staff asked the applicant to justify the elimination of the 103-percent trip.

In response to RAI-SRP10.2-SBPA-03, in a letter dated July 31, 2008, the applicant stated that the 103-percent trip value previously provided in Table 10.2-2 in Revision 15 of the DCD was not for a turbine-trip condition. The described condition was for the speed-control mode of the turbine-control system for a load reject event and a generator breaker open condition. The applicant eliminated the 103-percent value from DCD Table 10.2-2 because the Toshiba turbine valves and hydraulic system are not designed to fully close at the 103-percent rated speed. The CVs are closed at approximately 105 percent of rated turbine speed, and the intercept valves are closed at approximately 107 percent of rated speed. However, before these overspeed points are reached, the CVs and intercept valves begin to close at 101 percent of the rated turbine speed, as stated in DCD Table 10.2-2. The system is designed to prevent the peak transient of 108 percent of rated speed from being exceeded, as stated in the table. The applicant stated that this had not changed from DCD Revision 15—as speed is reduced, the valves will reopen and modulate as needed to achieve and maintain 100-percent rated speed. The staff accepts the applicant's response to RAI-SRP10.2-SBPA-03, for the reasons explained above, as well as the the applicant's explanation that it had eliminated the 103-percent value from DCD Table 10.2-2 because of the new Toshiba design.

In the changes to Tier 1 Section 2.4.2, Table 2.4.2-2 of the AP1000 DCD, the applicant added the emergency electrical overspeed trip device as a component located in the turbine building. This component, in conjunction with the electrical overspeed trip device (which is already listed in Tier 1 Table 2.4.2-2) provides the diversity and redundancy for the turbine overspeed trip function that are specified by NUREG-0800 Section 10.2. The staff agrees that the emergency electrical overspeed trip device is an important design feature that should be included in the Tier 1 design description for the main turbine system. Therefore, the addition of the emergency electrical overspeed trip device as a component located in the turbine building in Tier 1 Table 2.4.2-2 is considered to be acceptable.

10.2.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for design certification (DC) met the requirements of Subpart B, "Standard Design Certifications," to 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the changes the applicant proposed to the AP1000 turbine generator in the AP1000 DCD. The staff finds that incorporating the proposed changes does not adversely affect the ability of the turbine generator to meet the applicable acceptance criteria in NUREG-0800 Section 10.2. The staff also finds that the applicant has properly incorporated the design changes into the appropriate sections of the AP1000 DCD. Because the AP1000 turbine-generator design continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 turbine generator are acceptable.

In addition, the changes establish the proposed Toshiba design as the single, standard design for all AP1000 plants. These DCD changes are generic and are expected to be included in all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes to the AP1000 DCD contribute to the increased standardization of the certification information in the AP1000 DCD.

10.2.5 Valve Control

10.2.5.1 Summary of Technical Information

NUREG-0800 Section 10.2, Criterion II.1.B, states that the applicant should provide turbine main steam stop and CVs and reheat steam stop and intercept valves to protect the turbine from exceeding set speeds, as well as to protect the reactor system from abnormal surges. To ensure that turbine overspeed is controlled within acceptable limits, the reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves (MSVs) or of sequential closure within an appropriate time limit. The valve arrangements and valve closure times should ensure that a failure of any single valve to close will not result in an excessive turbine overspeed in the event of a turbine-generator system trip signal.

10.2.5.2 Evaluation

The valve arrangement in the new design is basically the same as in the original design, with the following exception: The No. 2 and No. 4 stop valves in the new design have a bypass valve, which is controlled by an electrohydraulic servo actuator for CV warming. The closure time for all stop, control, reheat, and intercept valves is 0.3 seconds for both the new and the original design.

The specific criterion that applies to the change evaluated above is 10 CFR 52.63(a)(1)(vi) in that the change substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security.

10.2.5.3 Conclusion

On the basis of its review, the staff finds that the design change is acceptable, because the availability and adequacy of the CVs in the AP1000 design conform to Criterion II.1.B and Criterion II.3 in NUREG-0800 Section 10.2.

10.2.8 Turbine Rotor Integrity

10.2.8.1 Summary of Technical Information

In the proposed changes to the AP1000 DCD, the applicant proposed an alternative SPC design that includes changing the design of the turbine rotors from a Westinghouse/Mitsubishi model to a Toshiba model. This evaluation addresses only the turbine-rotor design change. Revision 0 of TR-86, dated February 8, 2007, provided the technical justification for the proposed changes. Revision 1 of TR-86, submitted on June 25, 2007, established the Toshiba design as the single, standard turbine-rotor design for all AP1000 plants. In a letter dated July 12, 2007, the applicant submitted additional information on the TR. The staff evaluates the rotor design and probability of missile generation for the Toshiba turbine design below.

10.2.8.2 Evaluation

GDC 4 requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff reviewed the AP1000 DCD changes related to this section that ensure turbine-rotor integrity and a low probability that turbine-rotor failure will result in the generation of missiles.

In the proposed changes to AP1000 DCD, the applicant proposed a change to the design of the turbine rotors from a Westinghouse/Mitsubishi model to a Toshiba model. The Toshiba design includes two types of bucket (blade) fixations instead of side-entry blades, as in the Westinghouse/Mitsubishi design. The two types of fixations are an outside dovetail and a fork-type dovetail. The outside dovetail will be used in every stage of the high-pressure turbine and the fork-type dovetail will be used in the last two stages of the low-pressure turbine.

As a result of the design change, the applicant performed a new analysis of the missile generation probability for fully integral rotors of the Toshiba turbine-generator design. Westinghouse Commercial Atomic Power (WCAP)-16650-P, Revision 0, “Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low-Pressure Turbines,” issued February 2007, includes the methodology and results of the analysis. This report evaluates the probability of missiles resulting from four failure mechanisms: ductile burst from destructive overspeed, high-cycle fatigue cracking, low-cycle fatigue cracking, and stress-corrosion cracking. The staff reviewed the methodology used in WCAP-16650-P and its applicability to the Toshiba design. The applicant’s analysis concluded that an inspection interval of up to 24 years was sufficient to satisfy the requirement that the probability of missile generation be less than 1.0×10^{-5} per year. The applicant included conservative assumptions in its analysis for parameters, such as the stress required for ductile burst, the probability for stress-corrosion crack initiation, the probability of rotor overspeed, and the number of startup and shutdown cycles. In RAI-SRP10.2.3-CIB1-01, the staff requested additional information about the way in which the new missile probability analysis addressed high-trajectory missiles.

In a letter dated June 20, 2008, the applicant stated that the AP1000 design has a favorable turbine orientation that prevents turbine missiles from causing unacceptable damage to safety-related SSCs. The analysis in WCAP-16650-P demonstrates that for high-trajectory missiles, the probability of generating a missile from a burst turbine rotor (P1) is 10^{-5}. This meets the guidance in NUREG-0800 Section 3.5.1.3 of 10^{-4} for a favorable turbine orientation and 10^{-5} for an unfavorable turbine orientation. The staff finds this acceptable, because the the applicant’s analysis for high-trajectory missiles meets the guidance in NUREG-0800 Section 3.5.1.3.

The applicant also stated that the analysis in WCAP-16650-P applies only to high-trajectory missiles and that high-velocity, low-trajectory missiles cannot directly strike safety-related systems and components. However, the turbine generators at dual-unit sites must be considered unfavorably oriented, because safety-related systems and components are in the low-trajectory missile strike zone. Therefore, to address COL applications for dual-unit sites, the staff requested that the applicant provide a bounding turbine missile analysis for low-trajectory missiles or provide a COL action item that requires COL applicants to provide a turbine missile analysis for low-trajectory missiles at dual-unit sites. In a letter dated April 13, 2009, the applicant stated that WCAP-16650 also applies to low-trajectory turbine missiles for dual unit sites that have unfavorable orientation. However, the applicant did not provide justification for applying this analysis for low-trajectory turbine missiles, and did not address whether the analysis is dependent on missile trajectory. The staff identified this as Open Item OI-SRP10.2.3-CIB1-01.

In a letter dated September 22, 2009, the applicant provided further clarification of the applicability of WCAP-16650. The applicant stated that WCAP-16650-P determines the probability of generating a missile due to a burst turbine rotor, regardless of whether the missile is high-trajectory or low-trajectory. In addition, the applicant stated that the AP1000 DCD does not consider low-trajectory missiles for a single unit site. However, if a dual unit site is considered, the probability, P1, from WCAP-16650-P can be used to evaluate low-trajectory missiles. Therefore, the staff finds that since the angle of trajectory is not used in WCAP-16650-P for determining the total probability of generating a missile due to a burst turbine rotor, P1, the results of the analysis can be used for both low and high-trajectory missiles. The P1 result from WCAP-16650-P is then used to ensure that the probability of striking SSCs, P4, is less than 10^{-7} per the guidance of regulatory guide (RG) 1.115, “Protection Against Low-Trajectory Turbine Missiles,” Revision 1, and NUREG-0800 Section 3.5.1.3. On this basis, the staff finds the response acceptable; therefore, Open Item OI-SRP10.2.3-CIB1-01 is closed.

Based on the conservative assumptions incorporated into this analysis and the fact that the rotors will be inspected during plant operation at approximately 10-year intervals, the staff finds the methodology and results acceptable. The staff, therefore, finds the proposed rotor design changes acceptable, because they do not change the probability of missile generation for fully integral low-pressure rotors previously accepted by the NRC staff in NUREG-1793.

The staff reviewed the proposed changes to the AP1000 DCD. The changes establish the proposed design as the single, standard design for all AP1000 plants. These DCD changes are generic and are expected to be included in all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD.

10.2.8.3 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable, because it meets the requirements of GDC 4 and does not change the probability of missile generation for fully integral rotors that the staff previously accepted. Furthermore, the staff finds that the conclusions regarding the proposed turbine-rotor design and the methodology for analyzing the probability of missile generation are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable, under 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the DC information.

10.2.10 Valve Testing Intervals

10.2.10.1 Summary of Technical Information

Section 10.2.3.6 of the AP1000 DCD, proposed a decrease in the frequency of turbine-valve testing from every 3 months to every 6 months. To support this change, the applicant submitted TR WCAP-16651-P, Revision 0, “Probabilistic Evaluation of Turbine Valve Test Frequency,” issued February 2007. This report is similar to WCAP-15785, issued April 2002, with the same title, which used the same methodology to evaluate turbine-valve test frequency for the Westinghouse/Mitsubishi turbine design approved in Revision 15 of the AP1000 DCD. The methodology in WCAP-16651-P is based on an analysis of the operating experience with Toshiba Corporation nuclear steam turbines. The applicant introduced a proposed change to the DCD by replacing the word “quarterly” with “six-month” or “semi-annual” as follows:

> Turbine valve testing is performed at six-month intervals. The semi-annual testing frequency is based on nuclear industry experience that turbine-related

tests are the most common cause of plant trips at power. Plant trips at power may lead to challenges of the safety-related systems. Evaluations show that the probability of turbine missile generation with a semi-annual valve test is less than the evaluation criteria.

10.2.10.2 Evaluation

For the plant-specific turbine-rotor test data, the current COL Information Item 10.2-1 states that the COL holder "will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis." The applicant proposed to revise this information item by adding, to the end, the following time commitment: "prior to fuel load after the fabrication of the turbine." TR-6, "AP1000 As-Built COL Information Items" (APP-GW-GLR-021), Revision 0, issued June 2006, stated that an applicant cannot provide the as-built data at the time of the COL application. This justification is appropriate because the as-built material data for turbine rotors will not be available until the rotor material is procured for fabrication. In addition, the proposal to provide the as-built data before fuel load after the fabrication of the turbine is acceptable for reasons similar to those allowing the deferral of the maintenance and inspection program.

Regarding the proposal to extend the valve-test interval from 3 months to 6 months, the analysis in WCAP-16651-P showed that, for a favorably oriented turbine, the missile-probability criterion of 10^{-4} per year could be met, based on the operating experience with Toshiba turbines. In response to RAI-SRP10.2.3-CIB1-02, the applicant confirmed that the AP1000 turbine has a favorable orientation with respect to high-trajectory missiles. The analysis also concluded that the turbine-missile probability criterion of 10^{-5} per year could be met with a 6-month valve-test interval. This is significant because, for dual-unit sites, the turbine orientation is considered unfavorable with respect to low-trajectory missiles potentially striking safety-related systems and components. In response to RAI-SRP10.2.3-CIB1-02, the applicant provided clarifying information about the calculations used to determine the probability of missile generation, based on the frequency of valve tests.

The analysis presented in WCAP-16651-P, along with clarifying information provided in response to RAI-SRP10.2.3-CIB1-02, indicates that the existing operating experience for Toshiba turbines supports a missile generation probability of less than 10^{-5} per year, using 95-percent confidence-level values for the system separation frequency and valve-failure frequency. Therefore, based on the information provided, the staff concludes that the 6-month valve testing frequency meets the missile generation acceptance criteria for both favorable orientation (10^{-4} per year) and unfavorable orientation (10^{-5} per year).

However, the staff notes that an apparent error exists in Table 6-5 of WCAP-16651-P, Revision 0. Based on the equation given on page 18 of WCAP-16651-P, Revision 0, the staff's calculations indicated that, for a 6-month valve-test interval, the value of "probability of a turbine missile" for "1/Time Interval" was an order of magnitude lower than the value shown in Table 6-5. The value in Table 6.5 of Revision 0 does not support a six-month valve test frequency, and the staff identified this as Open Item OI-SRP10.2.3-CIB1-02. In a letter dated July 10, 2009, the applicant addressed this open item by submitting WCAP-16651-P, Revision 1. The applicant confirmed that the exponent in the probability value in Table 6-5 of Revision 0 was a typographical error. The corrected value in Table 6-5 of Revision 1 is consistent with the applicant's calculated value of the annual probability of a turbine missile that was used to support the six-month valve test interval. Therefore, Open Item OI-SRP10.2.3-CIB1-02 was closed.

10.2.10.3 Conclusion

Based on this evaluation, the staff concludes that the DCD changes meet the requirements of GDC 4 and thus are acceptable. The proposed DCD changes are acceptable, under 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information.

10.2.11 Turbine Rotor Maintenance and Inspection Program

10.2.11.1 Summary of Technical Information

The proposed changes to the AP1000 DCD revised COL Information Item 10.2-1 to change the timing for the COL holder to provide information on the turbine-rotor maintenance and inspection program from within 3 years of obtaining a COL until prior to fuel load. NUREG-1793 also discusses this information item under COL Action Item 10.5-2. TR-6 justified the revision to COL Information Item 10.2-1. The DCD changes also proposed a decrease in the frequency of turbine-valve testing from every 3 months to every 6 months.

In AP1000 DCD Tier 2, Revision 15, Section 10.2.6, COL Information Item 10.2-1 states the following:

> The Combined License holder will submit to the staff for review and approval within 3 years of obtaining a Combined License, and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis.

In the proposed changes to the AP1000 DCD, the applicant proposed a modification to COL Information Item 10.2-1 by changing the COL holder commitments to “prior to fuel load.” The proposed revision to DCD Tier 2, Section 10.2.6, states the following:

> The Combined License holder will submit to the NRC staff for review prior to fuel load, and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis prior to fuel load after the fabrication of the turbine.

10.2.11.2 Evaluation

GDC 4 requires that SSCs important to safety be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff reviewed the AP1000 DCD changes related to this section that ensure turbine rotor integrity and a low probability of missile generation caused by turbine rotor failure.

COL Information Item 10.2-1 addressed two issues: the turbine maintenance and inspection program and the plant-specific turbine rotor test data. TR-6 proposed modifications to both commitments.

For the turbine maintenance and inspection program, TR-6 states the following:

> A turbine maintenance and inspection program that is consistent with the frequency in DCD Subsection 10.2.3 does not need further approval from the NRC. The testing and inspection frequency in DCD Subsection 10.2.3 are supported by evaluations that are based on operating and inspection program experience in operating power plants.

For the maintenance and inspection program, the applicant proposed that the current COL Information Item 10.2-1 replace the commitment “submit to the staff for review and approval within 3 years of obtaining a Combined License” with “submit to the NRC staff for review prior to fuel load.” With respect to the proposed change in valve-test frequency, the staff requested, in RAI-SRP10.2.3-CIB1-02, that the applicant clarify the analysis in WCAP-16651-P with respect to the evaluation criterion.

Since the deferral of the activities in the COL information item does not alter the design of the turbine, turbine valves, or connected piping systems, it will not affect turbine rotor integrity. Further, this change does not involve a test or experiment, does not affect design features associated with the mitigation of severe accidents, and does not alter barriers or alarms that affect security assessments of the AP1000. Therefore, the staff finds that the applicant revised COL Information Item 10.2-1 to be acceptable. This determination is also consistent with the acceptance criteria in NUREG-0800 Section 10.2.3, “Turbine Rotor Integrity.”

10.2.11.3 Conclusion

Based on this evaluation, the staff concludes that the DCD changes meet the requirements of GDC 4 and thus are acceptable. Furthermore, the staff finds that the TR-6 conclusions regarding the submittal and availability of COL holder information for a turbine-rotor maintenance and inspection program are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable, under 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information.

10.3 Main Steam Supply System

10.3.1 Main Steam Supply System Design

10.3.1.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.3, “Main Steam Supply System (MSSS),” of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant proposed changes to this section, supported by the following Westinghouse reports: TR-86, “Alternate Steam and Power Conversion Design,” (APP-GW-GLN-018), Revision 1, issued June 25, 2007; TR-103, “Fluid System Changes” (APP-GW-GLN-019), Revision 2, dated January 29, 2008; and TR-125, “Corrections to Tier 1 ITAAC 2.2.4 and Tier 2 Section 3.6.1.3.3 and 10.3” (APP-GW-GLR-125), Revision 0, issued May 2007.

10.3.1.2 Evaluation

The staff reviewed the changes to AP1000 DCD, Section 10.3. NUREG-1793 includes the regulatory basis for Section 10.3 of the AP1000 DCD. The staff reviewed the proposed changes to DCD Section 10.3 against the applicable acceptance criteria in NUREG-0800 Section 10.3, “Main Steam Supply System.” Those changes that involve NRC review considerations as reflected in NUREG-0800 Section 10.3 are described and evaluated in this section. The specific criteria that apply to the MSSS design related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information. The results of the staff’s evaluation are provided as follows:

(1) Section 10.3.2.2.4, “Main Steam Isolation Valves,” Page 10.3-8:

The staff reviewed the applicant’s additional language stating that, on loss of electric power, the main steam isolation valves (MSIVs) remain in their current position. Since the closure of the MSIVs is credited for the mitigation of several design-basis transients and accidents, such as an inadvertent opening of a steam generator relief or safety valve, steam system piping failure, or steam generator tube rupture, the staff asked the applicant, in RAI-SRP10.3-SBPA-01, to evaluate the effect that maintaining the MSIVs in an open position after a loss of electrical power would have on the mitigation of these transients and accidents. Also, the staff asked the applicant to explain whether it had analyzed the events in DCD Chapter 15, “Accident Analyses,” with this MSIV fail-as-is logic.

In its response, dated July 18, 2008, the applicant clarified that the DCD statement, “On loss of electrical power the valves remain in their current position,” meant that the MSIVs would not close on loss of electric power to the actuator, since the solenoids are normally deenergized and closed. This is not to imply that the MSIVs will not close, once a steamline isolation signal is generated. As discussed in DCD Section 10.3.2.2.4, each MSIV is provided with a hydraulic/pneumatic actuator. The valve actuator consists of a hydraulic cylinder with a stored energy system to provide emergency closure of the MSIV. The energy to operate the valve actuator is stored in the form of compressed nitrogen contained in one end of the actuator cylinder. The MSIVs are normally maintained in an open position by high-pressure hydraulic fluid. For emergency valve closure, normally deenergized and closed redundant solenoids for each MSIV are energized, resulting in the high-pressure hydraulic fluid being dumped to a fluid reservoir and the compressed nitrogen closing the MSIV. Additionally, the applicant stated that the redundant solenoids for each MSIV are powered by separate divisions from the Class 1E direct current and uninterrupted power supply systems. Energizing either solenoid for each MSIV will close the MSIV. The loss of both redundant power sources is beyond the design basis of the plant.

Based on the above explanation, the staff finds the the applicant’s response acceptable for the following reasons:

- As described above, high-pressure hydraulic fluid in the actuators maintains the MSIVs open and, therefore, a loss of power will have no effect on the MSIV position. In addition, since the solenoids are normally deenergized, a loss of power will have no effect on their ability to keep the MSIVs open.

- To close the MSIVs during an emergency, the redundant solenoids powered by separate power sources will be used to dump the high-pressure fluid from the actuators. Since the solenoids are provided by two separate sources of power, the loss of an electric power source should have no effect on the MSIV position. Therefore, the staff finds the applicant's response acceptable, and the concern described in RAI-SRP10.3-SBPA-01 is resolved.

(2) Table 10.3.2-1, "Main Steam Supply System Design Data," Page 10.3-17:

The applicant modified the steam generator flow rates to conform to the new SPC design described in TR-86, Revision 1. In DCD Revision 15, the steamflow rate per steam generator was 3,396,535 kilograms per hour (kg/h) (7,488,000 pounds per hour (lb/h)) and the total steamflow rate was 6,793,069 kg/h (14,976,000 lb/h). In the revised DCD, the applicant changed the steamflow rate per steam generator to 3.40×10^6 kg/h (7.49×10^6 lb/h) and the total steamflow rate to 6.79×10^6 kg/h (14.97×10^6 lb/h). The staff reviewed these changes and determined that they are not significant and that they continue to meet all the acceptance criteria in NUREG-1793. Therefore, the staff considers these changes to be acceptable.

(3) Table 10.3.2-2, "Design Data for Main Steam Safety Valves," Page 10.3-18:

The staff reviewed the applicant's revisions to the set pressures and relieving capacities of the main steam safety valves (MSSVs). Since the MSSVs are credited for the mitigation of many design-basis transients and accidents, including overpressure protection of the heatup events, such as a loss of external electrical load and a turbine trip, the staff asked the applicant, in RAI-SRP10.3-SBPA-02, to provide additional information with respect to the effects of the revised MSSV set pressures and relieving capacities on the event analysis in Chapter 15. The staff also asked whether the applicant had performed the Chapter 15 event analysis with the revised MSSV setpoints and relieving capacities.

In its response, dated July 18, 2008, the applicant stated that it included the revised MSSV setpoints and capacities, as shown in Table 10.3.2-2 of the DCD, in its evaluation of the limiting Chapter 15 event analyses that were provided in its response to RAI-TR29-SRSB-01. The results of the evaluation provided in the response to RAI-TR29-SRSB-01 show that the effect of the changes to the MSSV parameters meets the acceptance criteria of the limiting Chapter 15 events and that the existing analysis is bounding.

Based on its review of the Chapter 15 event analyses, the staff finds that the applicant did include the revised MSSV setpoints and capacities in its evaluation of the limiting Chapter 15 event analyses. In addition, based on the results of the Chapter 15 event evaluation, the staff finds that the changes to the MSSV parameters meet the acceptance criteria of the limiting Chapter 15 events. Because the applicant performed the Chapter 15 event analyses with the revised MSSV setpoint pressures and relieving capacities and because the results remained within the acceptance criteria of the limiting Chapter 15 events, the staff's concern described in RAI-SRP10.3-SBPA-02 is resolved.

(4) Main Steam Piping and Instrumentation Diagrams (Safety-Related System), Pages 10.3-37 and 10.3-39 (Figure 10.3.2-1, two sheets):

The applicant revised the AP1000 DCD Tier 2, Figure 10.3.2-1, to be consistent with the applicant's proposed changes to DCD Tier 1, Section 2.2.4, "Steam Generator System," described in TR-125. This change pertains to the safety-related portion of the MSSS.

During an April 1, 2009, conference call, the staff asked the applicant to provide legible DCD Tier 2, Figures 10.3.2-1 and 10.3.2-2, and to incorporate the July 3, 2008, response to RAI-SRP 3.6.1-SBPA-01 into DCD Section 10.3.1.1. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD, which resolves this issue.

(5) Main Steam Piping and Instrumentation Diagrams (Safety-Related System), Pages 10.3-37 and 10.3-39 (Figure 10.3.2-1, two sheets):

The applicant changed the system arrangement and configuration design in DCD Tier 2, Figure 10.3.2-1, to conform to TR-103.

During an April 1, 2009, conference call, the staff asked the applicant to provide legible Tier 2, Figures 10.3.2-1 and 10.3.2-2 and to incorporate the July 3, 2008, response to RAI-SRP3.6.1-SBPA-01 into DCD Section 10.3.1.1 In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD, which resolves this issue.

(6) Main Steam System Diagram, Page 10.3-41 (Figure 10.3.2-2):

The applicant modified the MSSS arrangement to conform to the new SPC design, as proposed in TR-86, Revision 1. The applicant's changes to sections, tables, and figures of the DCD support the specifics of the new design. The staff reviewed these changes and finds that they are modifications to the system arrangement and configuration design. The staff finds that these changes will have no impact on the ability of the MSSS to supply steam from the steam generator to the main turbine-generator system. The staff finds that the MSSS will continue to meet all of the acceptance criteria in NUREG-1793 and, therefore, the changes are acceptable.

(7) a) Section 10.3.1.1, "Safety Design Basis," Page 10.3-3:

The applicant added the turbine bypass valve and also changed the moisture separator reheater stop valves to moisture separator reheater 2nd stage steam isolation valves in the list of valves credited in a single failure analysis. These valves are credited in the analysis to mitigate the event for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event. These two changes conform to the applicant's response to RAI-SRP3.6.1-SBPA-01, which was found acceptable in Section 3.6.1 of this report.

b) Tier 1, Table 2.2.4-3, "Components which Provide Backup Isolation of the Steam Generator System," Page 2.2.4-15:

The applicant added two moisture separator reheater 2nd stage steam isolation valves to the list of components, which provide backup isolation of the steam generator system and deleted other components. The staff reviewed this design change and determined that it was made to be consistent with the change made in a) above. Therefore, the staff finds the change to be acceptable.

c) Section 10.3.4.2, "In-service Testing," Page 10.3-12:

The applicant revised the scope of the inservice testing (IST) program for the AP1000 reactor to replace the moisture separator reheater steam supply control valves, MSS-PL-V016A, MSS-PL-V016B, MSS-PL-V017A, and MSS-PL-V017B, with the moisture separator reheater 2nd stage steam isolation valves, MSS-PL-V015A and MSS-PL-V015B, in response to the design change described in a) above. The staff reviewed the revised IST program description for consistency with the design change. MSS-PL-V015A and MSS-PL-V015B have been included in AP1000 DCD Tier 2, Table 3.9-16, because the design change will have those valves perform a backup isolation function for the safety-related MSIVs in the AP1000 reactor design rather than MSS-PL-V016A, MSS-PL-V016B, MSS-PL-V017A, and MSS-PL-V017B. MSS-PL-V015A and MSS-PL-V015B are nonsafety related valves (without a specific safety-related leakage limitation) that are outside the scope of the IST requirements specified in 10 CFR 50.55a, "Codes and standards," and will be included in the IST program for the AP1000 reactor to augment their performance capability. Based on its review, the staff finds the revision to the IST program scope to be consistent with the design change described in a) above.

(8) Piping Material Changes to Section 10.3.2.2.1, "Main Steam Piping," Page 10.3-5; Table 10.3.2-3, "Description of Main Steam and Main Feedwater Piping," Page 10.3-19; Main Steam Piping and Instrumentation Diagrams (Safety-Related System) and Page 10.3-37 (Figure 10.3.2-1, "Main Steam Piping and Instrumentation Diagram (Safety-Related System) – Sheet 1 of 2):

The applicant changed the material used for the main steam piping and segments of the main feedwater line to low alloy steel to facilitate a design life of 60 years with respect to minimizing the effects of erosion/corrosion. In the certified design, carbon steel is the main steam piping material for some portions. The changes proposed in Revision 17 make P11 material (nominally 1.25 percent chromium and 0.5 percent molybdenum) the only material for main steam piping. The staff finds this material is acceptable because testing and operating experience have shown P11 is resistant to flow-accelerated corrosion (sometimes called erosion-corrosion) in main steam piping, including wet steam. Current industry guidance (Electric Power Research Institute (EPRI) NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program") allows P11 main steam components to be excluded from flow-accelerated corrosion inspection programs due to the flow accelerated corrosion resistance imparted by the chromium and, to a lesser extent, molybdenum.

(9) Section 10.3.3, "Safety Evaluation," Page 10.3-10:

The applicant changed from inline main steam line radiation monitors to adjacent-to-line radiation monitors.

The applicant made the change to Section 10.3.3 to be consistent with the detailed monitor description provided in DCD Section 11.5.2.3.1. DCD Section 11.5.2.3.1 has not changed from the DCD Revision 15 description of these monitors as "…adjacent to the steam lines." The staff concludes that both DCD sections are now consistent and that the change to Section 10.3 is acceptable. The staff provides additional evaluation of the main steam line monitors in Section 11.5.2.4 of this report.

(10) Table 10.3.2-2, "Design Data for Main Steam Safety Valves," Page 10.3-18:

In Table 10.3.2-2, the applicant changed the size of MSSV and added clarifying text. The MSSVs are designed to American Society of Mechanical Engineers (ASME) Code Section III, Class 2, seismic Category I. Based on the system's accumulation pressure of 3 percent, in accordance with Subsection NC-7512 of ASME Code, Section III, Division 1, 1989 Edition, Subsection NC, Class 2 components, the applicant increased the relieving capacity of MSSVs. The changes were made to ensure that the MSSV design meets the requirements and conforms to the ASME Code. The staff finds the changes acceptable.

(11) Table 10.3.2-4, "Main Steam Branch Piping (2.5-Inch and Larger) Downstream of MSIV," Page 10.3-20:

The applicant changed, in DCD Table 10.3.2-4: the size of the shutoff valves for the turbine bypass lines to the condenser from 30.5 cm (12 in) to 40.6 cm (16 in), the size of the shutoff valves for the reheating steam to moisture separator reheater lines from 30.5 cm (12 in) to 25.4 cm (10 in), and the numerical value for the maximum steam flow rate for these lines. The description of the shutoff valve and number of lines under the description column was also changed. Further, the size of the shutoff valve for the main steam supply to auxiliary steam system line was changed from 15.2 cm (6 in) to 25.4 cm (10 in).

As reflected in Table 10.3.2-4, the applicant made changes to the main steam branch piping, in order to meet the analyzed closure time for the MSIV backup isolation valves.

The applicant made changes to the main steam shutoff valves in Table 10.3.2-4 to support the change in the turbine generator design and to optimize the auxiliary steam supply system. The staff reviewed these design changes and determined that they are bounded by existing safety analysis because the shutoff valve closure times remain unchanged. The design will continue to meet all acceptance criteria in NUREG-1793. The staff finds the changes are acceptable.

10.3.1.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 MSSS. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 MSSS to meet the applicable acceptance criteria in NUREG-0800 Section 10.3. The staff also finds that the applicant properly incorporated the design changes into the appropriate sections of AP1000 DCD. Because the AP1000 MSSS design continues to meet all of the applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 MSSS are acceptable.

10.4 Other Features

10.4.1 Main Condensers

10.4.1.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.1, "Main Condenser," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant proposed to make the following changes to Section 10.4.1 of the DCD.

In DCD Section 10.4.1.2, "System Description," Paragraph 2, the applicant revised the condenser tube material providing an option to the COL applicant to substitute "stainless steel" tubes for "titanium" tubes for fresh water cooled plants. The revised description reads as, "…..The condenser is equipped with titanium <u>or stainless</u> steel tubes. The titanium material provides good corrosion and erosion resisting properties. <u>Fresh water cooled plants do not require the high level corrosion and erosion resistance provided by titanium; therefore, 304L, 316L, 904L, or AL-6X may be substituted if desired.</u>" The revisions are indicated by the underlined text. The applicant added a corresponding note to Table 10.4.1-1, "Main Condenser Design Data." In addition, a second note added to Table 10.4.1-1 states that if one of the alternate tube materials (i.e., stainless steel) is used, the tubesheet will be carbon steel material with a cladding of the same material as the tubes. The certified design specified titanium or titanium-clad carbon steel as the tubesheet material.

10.4.1.2 Evaluation

The staff reviewed the proposed changes to AP1000 DCD, Section 10.4.1. The regulatory basis for AP1000 DCD, Section 10.4.1, is documented in NUREG-1793. The staff has reviewed the proposed changes to DCD Section 10.4.1 against the applicable acceptance criteria of NUREG-0800 Section 10.4.1, "Main Condensers."

The changes identified above are related to the material of the condenser tubes. Several stainless steel alloys (304L, 316L, 904L, and AL-6X) were added as alternatives to titanium condenser tube material for plants cooled by fresh water. The staff finds the use of stainless steel materials is acceptable because the corrosion of stainless steel in cooling waters is strongly related to the chloride content. Operating experience has shown that the stainless steels proposed by the applicant are suitable for condenser tubes in fresh water applications. For example, EPRI characterizes the corrosion resistance of these stainless steels as very good to excellent in fresh water (EPRI TR-102922, "High-Reliability Condenser Application Study," Final Report, November 1993). The corresponding change to allow the tubesheet to be carbon steel with a cladding of the same stainless steel as the tube material is also acceptable based on the operating experience referenced above.

10.4.1.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed Section 10.4.1, "Main Condensers," as documented in the AP1000 DCD, Revision 17. On the basis of its review, the staff finds that the design changes have been adequately incorporated into the appropriate sections of the AP1000 DCD, Revision 17. The staff also finds, based on operating experience with stainless steel condenser tubes, that the main condenser continues to meet all acceptance criteria as documented in NUREG-1793 and are, therefore, considered to be acceptable.

10.4.2 Main Condenser Evacuation System

10.4.2.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.2, "Main Condenser Evacuation System," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant revised Sections 10.4.2.2.1, "General Description"; and 10.4.2.2.2, "Component Description," to identify the circulating water system (CWS) as conceptual design information (CDI).

10.4.2.2 Evaluation

The staff reviewed all of the changes to AP1000 DCD, Section 10.4.2. The staff did not re-review descriptions and evaluations of Section 10.4.2 that were previously approved and that are not affected by the new changes.

The specific criterion that applies to the main condenser evacuation system related changes is 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.2 of the AP1000 DCD. The staff has reviewed the proposed changes to DCD Section 10.4.2 against the applicable acceptance criteria in NUREG-0800 Section 10.4.2.

The only change the applicant proposed to AP1000 DCD, Section 10.4.2, was to identify the CWS as CDI. The staff discusses the issue of identifying the CWS as CDI in Section 10.4.5 of this report. Since the changes made to DCD Sections 10.4.2.2.1 and 10.4.2.2.2 conform to the changes made to DCD Section 10.4.5 and do not adversely affect the main condenser evacuation system, the staff finds these changes to be acceptable.

10.4.2.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusion that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the proposed changes to AP1000 DCD, Section 10.4.2. The staff finds no changes except the two conforming changes identified above. On the basis of its review, the staff finds that the main condenser evacuation system continues to meet all of the acceptance criteria in NUREG-1793, and these changes are, therefore, acceptable.

10.4.3 Turbine Gland Seal System

10.4.3.1 Summary of Technical Information

In NUREG-1793, the staff approved AP1000 DCD, Revision 15, Section 10.4.3, "Gland Seal System." In the proposed changes to the AP1000 DCD, the applicant proposed changes to Section 10.4.3 of the DCD.

In AP1000 DCD, Section 10.4.3, the applicant made the following changes, which reflect the new single turbine-generator design (i.e., alternative SPC design) that is proposed in TR-86, Revision 1:

(1) Section 10.4.3.2.2, "System Operation," of the DCD, Revision 16:

AP1000 DCD, Revision 15, Section 10.4.3.2.2, stated that the sealing steam to the turbine shaft seals is supplied from either the auxiliary steam system or from the main steam system extracted ahead of the high-pressure turbine throttle valve. In the DCD Revision 17, the applicant revised Section 10.4.3.2.2 to refer to the "high-pressure turbine control valve."

In addition, DCD Section 10.4.3.2.2, Revision 15, stated that, at times other than initial startup, turbine-generator sealing steam is supplied from the auxiliary steam system or from main steam. In the DCD, Revision 17, the applicant modified paragraph 3 on page 10.4-7 to add "the MSV (main steam stop valve) and CV (control valve) gland steam leak-off" as an additional source of steam to the turbine gland seals.

(2) Section 10.4.3.5, "Instrumentation Applications," of the DCD, Revision 17:

The applicant removed the following sentence: "Pressure control valves are used to provide appropriate pressures to operate both the low and high pressure turbine steam seals."

10.4.3.2 Evaluation

The staff reviewed all of the proposed changes to AP1000 DCD, Section 10.4.3. The staff did not review descriptions and evaluations of Section 10.4.3 that were previously approved and that are not affected by the new changes.

NUREG-1793 includes the regulatory basis for Section 10.4.3 of the AP1000 DCD. The staff has reviewed the proposed changes to AP1000 DCD, Section 10.4.3, against the applicable acceptance criteria of NUREG-0800 Section 10.4.3, "Turbine Gland Sealing System."

The specific criterion that applies to the turbine gland seal system related changes is 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

The staff finds that the changes to DCD Section 10.4.3.2.2 associated with specifying a high-pressure turbine CV and the addition of sealing steam supply from MSV and CV gland steam leakoff are conforming changes to the new design, in accordance with TR-86, Revision 1, and do not affect the staff's conclusion in NUREG-1793 that the gland seal system design meets the requirements of GDC 60, "Control of Releases of Radioactive Materials to the Environment," with respect to the design features in place to control releases of radioactive materials to the environment. Therefore, the staff finds these changes acceptable.

However, with respect to the removal of the pressure CVs from DCD Section 10.4.3.5, "Instrumentation Application," the applicant did not provide a basis for this change. Therefore, in RAI-SRP10.4.3-SBPA-01, the staff asked the applicant to explain and justify this deletion of the pressure-regulating valves.

In response to RAI-SRP10.4.3-SBPA-01, in its letter dated June 20, 2008, the applicant stated that the AP1000 gland seal system uses pressure-regulating valves to control steam pressure to the turbine glands. In the approved design, as described in the DCD, Revision 15, the high-pressure and low-pressure turbine glands used steam at different pressures and, therefore, had their own pressure-regulating valves to deliver their respective pressures. In the proposed changes to the DCD; however, the applicant changed its design to a single-pressure system, in which the high- and low-pressure turbine glands use the same steam pressure from a common gland seal steam supply header. Figure 10.4.3-1 of the DCD, Revision 17, includes the diagram depicting the gland seal steam piping and instrumentation. The gland seal steam header is supplied with steam from both the auxiliary steam and main steam systems, and the above-cited pressure-regulating valves are located in each of these systems upstream of the header. Based on the above discussion and a review of Figure 10.4.3-1, the staff finds the elimination of the gland seal steam dual-pressure system acceptable, since it is the new Westinghouse design and does not affect any safety-related systems or equipment. Therefore, the staff's concern described in RAI-SRP10.4.3-SBPA-01 is resolved.

10.4.3.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the proposed changes to Section 10.4.3of the AP1000 DCD. The staff finds that the changes with respect to specifying a high-pressure turbine CV, the addition of seal steam supply from MSV and CV gland steam leakoff, and other changes to the gland seal system design proposed in TR-86, Revision 1, do not affect the staff's conclusion in NUREG-1793 that the gland seal system design meets the requirements of GDC 60, with respect to the design features in place to control releases of radioactive materials to the environment. The proposed changes do not adversely affect the AP1000 turbine gland seal steam system.

10.4.4 Turbine Bypass System

10.4.4.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.4, "Turbine Bypass System," of the AP1000 DCD, Revision 15.

In the proposed changes to the AP1000 DCD, the applicant proposed to clarify Section 10.4.4.2.2, "Component Description," regarding the manual use of the bypass valves during cooldown. The revised sentence reads: "…the low T_{avg} block can be manually bypassed for ~~one of~~ the bypass valves <u>that are designated as cooldown valves</u> to allow operation during plant cooldown." The addition is indicated by underlined text.

10.4.4.2 Evaluation

The staff reviewed the proposed changes to Section 10.4.4. The specific criterion that applies to the change is 10 CFR 52.63(a)(1)(vii), in that the proposed change contribute to increased standardization of the certification information.

The regulatory basis for AP1000 DCD, Section 10.4.4, is documented in NUREG-1793. The staff reviewed the proposed changes to DCD Section 10.4.4 against the applicable acceptance criteria of NUREG-0800 Section 10.4.4. The acceptability of the system design is based on meeting the following GDC as described in NUREG-0800:

- GDC 4, as it relates to the system being designated such that a failure of the system (due to a pipe break or system malfunction) does not adversely affect safety-related systems or components

- GDC 34, "Residual Heat Removal," as it relates to the ability to use the turbine bypass system for shutting down the plant during normal operations by removing residual heat without using the turbine generator

The change proposed by the applicant is considered a clarification and does not alter the ability of the system to manually bypass a T_{avg} block and allow operation of the bypass valves during plant cooldown. Therefore, the change is considered acceptable.

10.4.4.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and thatthe application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the proposed changes to Section 10.4.4, "Turbine Bypass System." On the basis of its review, the staff finds the turbine bypass system continues to meet all acceptance criteria as documented in NUREG-1793 and is, therefore, considered to be acceptable.

10.4.5 Circulating Water System

10.4.5.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.5, "Circulating Water System," of the AP1000 DCD, Revision 15. In the proposed changes to the AP1000 DCD, the applicant proposed the following changes to Section 10.4.3.

In Section 10.4.5, the applicant identified CDI items related to the alternative SPC design proposed in TR-86, Revision 1. Additionally, as part of identifying the CDI information, the applicant revised Section 10.4.5 to include makeup water as an additional source to the CWS by stating the following:

> The CWS and/or makeup water from the raw water system supplies cooling water to the turbine building closed cooling water system (TCS) heat exchangers and the condenser vacuum pump seal water heat exchangers under varying conditions of power plant loading and design weather conditions.

The underlined text indicates the revision.

10.4.5.2 Evaluation

The staff reviewed the proposed changes to the CWS in the AP1000 DCD. The staff did not review descriptions and evaluations of Section 10.4.5 that were previously approved and that are not affected by the new changes. The specific criterion that applies to the changes is 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.5 of the AP1000 DCD. The staff has reviewed the proposed changes to DCD Section 10.4.5 against the applicable acceptance criteria of NUREG-0800 Section 10.4.5, “Circulating Water System,” and other applicable criteria.

Since the CWS system configuration and its components, such as circulating water pumps, cooling tower or heat sink, and piping and valves, are plant-specific items, the applicant identified these as CDI items. Further, the addition of makeup water to the CWS for the supply of cooling water to the turbine control system heat exchangers and the condenser vacuum pump seal water heat exchangers are plant-specific CDI items.

Therefore, the staff finds that the addition of makeup water and the identification of CDI items are changes to conform to the new design, in accordance with TR-86, Revision 1, and do not affect the staff’s conclusion in NUREG-1793 that the CWS design meets the requirements of GDC 4 with respect to the effects of discharging water that may result from a failure of a component or piping in the CWS. Therefore, the staff finds these changes to be acceptable.

10.4.5.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the proposed changes to Section 10.4.5. The staff finds that the addition of the makeup water system and the identification of the CDI items are acceptable, as they relate to the new design proposed in TR-86, Revision 1. Further, these revisions to Section 10.4.5 of the DCD, do not adversely affect the ability of the CWS to meet the requirements of GDC 4. On the basis of its review, the staff finds that the CWS continues to meet all of the acceptance criteria in NUREG-1793 and is, therefore, acceptable.

10.4.7 Condensate and Feedwater System

10.4.7.1 Summary of Technical Information

In NUREG-1793, the staff approved Section 10.4.7, "Condensate and Feedwater System," of the AP1000 DCD, Revision 15. The applicant proposed changes to Section 10.4.7 of the AP1000 DCD.

The applicant proposed the following four technical changes to the AP1000 DCD, three of which are supported by information in Westinghouse TRs:

(1) In Revision 17 to the AP1000 DCD, pages 10.4-21, 10.4-22, 10.4-26, and drawing page 10.4-71 (Figure 10.4.7-1, Sheet 3 of 4), the applicant proposed changing the number of high-pressure feedwater heater stages from one to two. The AP1000 DCD, Revision 15, designates the one high-pressure feedwater heater stage as feedwater heater number 6, with the stage consisting of two parallel feedwater heaters, numbers 6A and 6B. The AP1000 DCD, Revision 17, designates the additional stage feedwater heater number 7, with the stage consisting of two parallel feedwater heaters, numbers 7A and 7B. TR-86, Revision 1, documents the basis for this change.

(2) In the AP1000 DCD, Revision 17, page 10.4-25, the applicant changed text in the first and second paragraphs under the title "Low-Pressure Feedwater Heaters." The current text reads as follows:

> Except for the No. 1 feedwater heaters, the closed low-pressure feedwater heaters have integral drain coolers, and their shell side drains cascade to the next lower stage feedwater heater. The drains from the No. 1 heaters are dumped to their respective condenser shell.
>
> A drain line from each heater allows direct discharge of the heater drains to the condenser in the event the normal drain path is not available or flooding occurs in the heater.

The revised text reads as follows:

> The closed low-pressure feedwater heaters use drain coolers. The cascaded drains from the heaters are dumped to their respective condenser shell.
>
> A drain line from the heater allows direct discharge of the heater drains to the condenser in the event the normal drain path is not available or flooding occurs in the heater.

TR-86, Revision 1, documents the basis for this change. The applicant also incorporated this change in the AP1000 DCD, on drawing page 10.4-69 (Figure 10.4.7-1, Sheet 2 of 4), to show that the number 1 and number 2 feedwater heater drains dump into their respective condenser shells.

(3) In the AP1000 DCD, Revision 17, on page 10.4-73 (Figure 10.4.7-1, Sheet 4 of 4), the applicant proposed eliminating pressure transmitters PT043, PT044, and PT045, and

pressure instrument controllers PIC043, PIC044, and PIC045, and replacing them with PT042 and PC042. TR-103, Revision 2, documents this change.

(4) In the AP1000 DCD, Revision 17, on pages 10.4-69 and 10.4-71 (Figure 10.4.7-1, Sheets 2 and 3), the applicant apparently made a change to the arrangement of the deaerator feedwater heater and its associated storage tank.

10.4.7.2 Evaluation

The staff reviewed all changes to the condensate and feedwater system (CFS) in the AP1000 DCD. The staff did not re-review descriptions and evaluations of Section 10.4.7 that were previously approved and are not affected by the new changes. The specific criteria that apply to the CFS related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.7 of the AP1000 DCD. The staff has reviewed the proposed changes to DCD Section 10.4.7 against the applicable acceptance criteria of NUREG-0800 Section 10.4.7, "Condensate and Feedwater System." The following evaluation discusses the results of the staff's review.

10.4.7.2.1 Condensate and Feedwater System—Addition of High-Pressure Feedwater Heater Stage 7

In TR-86, Revision 1, Section 1, the applicant stated that an alternative SPC design was presented in Revision 0 of TR-86 as DCD Chapter 10A, and that all standard AP1000 units built will now employ a single turbine-generator and steam-cycle design. The applicant proposed incorporating changes from DCD Chapter 10A as a single design for AP1000 and stated that one of the significant differences between the reference design and the alternative new design of the turbine-generator set is the addition of a seventh-stage feedwater heater. Because the alternative design of the turbine-generator set uses a seventh-stage high-pressure feedwater heater in its feedwater system design, and because the addition of the seventh-stage heater has no effect on CFS compliance with GDC 2, "Design Bases for Protection against Natural Phenomena"; GDC 4; GDC 44, "Cooling Water"; GDC 45, "Inspection of Cooling Water System"; and GDC 46, "Testing of Cooling Water System," the staff finds the addition of the second high-pressure feedwater heater to the AP1000 CFS design acceptable. The staff also finds that the addition of the seventh-stage heater does not result in a change to the CFS compliance with NUREG-0800 Section 10.4.7 in NUREG-1793.

10.4.7.2.2 Condensate and Feedwater System, Low-Pressure Feedwater Heater Stages 1 and 2 Drain to Their Respective Condensers

In TR-86, Revision 1, on page 52, the applicant proposed a design change for low-pressure feedwater heater stages 1 and 2 drains to dump directly into their respective condenser shells. This design change is part of the alternative SPC design described in Revision 0 of TR-86 as DCD Chapter 10A. Also, on page 60 of TR-86, Revision 1, the applicant proposed a similar change to DCD Figure 10.4.7-1 (Sheet 2 of 4), which has been changed schematically to show that the drains from low-pressure feedwater heaters 1A, 1B, 1C, 2A, 2B, and 2C dump into the respective condenser shells A, B, and C. The low-pressure feedwater heater stage 2 drain no longer cascades into feedwater heater stage 1. Because the alternative design of the turbine-generator set uses a different drainpath for the No. 2 low-pressure feedwater heater stage, which does not affect the safety-related functions of the CFS or its compliance with GDC 2, 4, 44, 45, and 46, the staff finds this design change to the AP1000 CFS acceptable. The staff also finds that the change to the heater drain system does not result in a change to the CFS compliance with NUREG-0800 Section 10.4.7 in NUREG-1793.

10.4.7.2.3 Condensate and Feedwater System Pressure Transmitters PT043, PT044, and PT045, and Pressure Instrument Controllers PIC043, PIC044, and PIC045 Replaced with PT042 and PC042

In TR-103, Revision 2, on page 7, the applicant proposed eliminating the control function shown in Figure 10.4.7-1 (Sheet 4 of 4) for PT-043–45, because the startup feedwater header pressure is no longer an input to the startup feedwater control system. In TR-103, the applicant further stated that all three pressure transmitters are no longer necessary, since they do not have a control function and pressure transmitters PT-044 and PT-045 have been deleted. Additionally, the applicant has revised Note 4 to reflect the updated control logic diagram APP-PLS-J1-143 and has deleted the statement in the note that mentions startup feedwater header pressure as an input to the control system. The applicant also made a change in Note 3 to indicate that control logic diagram APP-PLS-J1-114 has been updated and is now APP-PLS-J1-144.

Because the startup feedwater header pressure signal no longer provides input to the startup feedwater control system, and the new control logic does not affect the CFS compliance with GDC 2, 4, 44, 45, and 46, the staff finds this design change to the AP1000 CFS acceptable. Section 7.7.1.5 of this report reviews the acceptability of instrumentation and controls changes related to AP1000 DCD Sections 7.7.1.8.1, “Feedwater Control”; and 7.7.1.8.2, “Startup Feedwater Control.”

10.4.7.2.4 Figure 10.4.7-1 (Sheets 2 and 3) Revised To Show the ME 05 “Deaerator Feedwater Heater and Deaerator Storage Tank” as a Single Component

The approved DCD shows the deaerator and the deaerator storage tank as separate components. However, AP1000 DCD, Revision 16, Figure 10.4.7-1 (Sheets 2 and 3), shows the deaerator feedwater heater and the deaerator storage tank as a single component. Furthermore, Section 10.4.7.2.2, “Component Description,” of the DCD describes the deaerator as “a tray type, horizontal shell, direct contact heater located on top of a horizontal storage tank.” Since the design of the deaerator and the deaerator storage tank in the AP1000 DCD, Revision 16, differs from the approved design and from the component description provided in the DCD, the staff, in RAI-SRP10.4.7-SBPA-01, dated April 28, 2008, asked the applicant to explain why the design change is acceptable.

In its response, dated June 20, 2008, the applicant stated that there was no change to the function of the deaerator or the feedwater and condensate systems, and the confusion was a result of the various descriptions and pictorial representations of the deaerator used throughout the DCD. A more consistent representation of the deaerator in the DCD would have likely prevented the confusion. The deaerator vendor changed the previous vendor’s design, which used a deaerator storage tank with the deaerating heater mounted on top of the tank. The latest vendor’s standard design is functionally the same, but the deaerating heater is mounted inside the deaerator storage tank.

The applicant further stated that the differences between the two vendors' deaerator designs do not have any effect on the design or function of the feedwater and condensate systems. The applicant changed DCD Figure 10.4.7-1 (Sheets 2 and 3), as well as the text in DCD Section 10.4.7.2.2, to more accurately reflect the current vendor's deaerator.

In evaluating the applicant's response, the staff notes that the applicant indicated that the deaerator vendor had changed and that in the new vendor's design, the deaerator heater is mounted inside the deaerator storage tank, as opposed to being located on top of the tank, as was the case in the DCD revisions before Revision 16. However, while the new design is shown in the revised Figure 10.4.7-1, the deaerator description in Section 10.4.7.2.2 continues to describe the dearator as "a tray type horizontal shell, direct contact heater located on top of a horizontal storage tank." The deareator described in Section 10.4.7.2.2 of the DCD, and that shown in Figure 10.4.7-1, are not consistent, despite the applicant's claim in its response that it changed the text in DCD Section 10.4.7.2.2 to more accurately reflect the current vendor's deaerator. The staff found the DCD information on the deaerator to be inconsistent and the applicant's response to be insufficient, since the response includes an inaccuracy regarding the text in Section 10.4.7.2.2. In DCD Revision 17, Section 10.4.7.2.2, the applicant revised the text that describes the deaerator to reflect what is shown in Figure 10.4.7-1 of the DCD. On the basis of its review, the staff finds RAI-SRP10.4.7-SBPA-01 to be resolved.

10.4.7.3 Conclusion

The staff reviewed the applicant's proposed changes to the CFS. The staff finds that the applicant's proposed changes do not affect the ability of the AP1000 CFS to meet the applicable acceptance criteria. The staff also finds that the applicant properly incorporated the design changes into the appropriate sections of the AP1000 DCD. Because the AP1000 CFS design continues to meet all of the applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff finds that all of the changes to the AP1000 CFS are acceptable.

10.4.8 Steam Generator Blowdown System

10.4.8.1 Summary of Technical Information

The staff approved Section 10.4.8, "Steam Generator Blowdown System," of the AP1000 DCD, Revision 15, in the certified design. In the proposed changes to the AP1000 DCD, the applicant revised Section 10.4.8 of the certified design.

- In DCD Section 10.4.8.2.2, the system flow rates were increased by approximately 8 to 9 percent. In a letter dated July 30, 2009, the applicant explained that the flow rates were changed in order to reflect updated system design calculations. In addition, the words, "at standard conditions" were added after the flow rates.

- Three wording corrections were made for consistency with other parts of the DCD.

- A note was added to Figure 10.4.8-1. The note identifies the drawing as a functional arrangement that could have different internal details due to procurement requirements.

10.4.8.2 Evaluation

The staff reviewed the changes to the steam generator blowdown system identified in Section 10.4.8 of the AP1000 DCD. The staff did not review descriptions and evaluations of Section 10.4.8 that were previously approved and that are not affected by the new changes. The specific criteria that apply to the steam and generator blowdown system related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.8 of the AP1000 DCD. The staff reviewed the proposed changes to the AP1000 DCD against the applicable acceptance criteria of NUREG-0800 Section 10.4.8, "Steam Generator Blowdown System," and other applicable criteria.

The staff finds the changes in flow rate values acceptable because they do not affect the amount of blowdown as a percentage of steam flow. Hence, the updated flow rates support the level of secondary-side water purification in the certified AP1000 design. Adding the phrase, "at standard conditions" is acceptable because it provides clarification to the flow-rate values.

The other wording changes were made for consistency with the design as documented in other parts of the DCD and are, thus, acceptable. The note added to Figure 10.4.8-1 is acceptable because it clarifies that there is flexibility in the as-built system without affecting the system functionality.

10.4.8.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 steam generator blowdown system. The staff finds that the applicant's proposed changes do not affect the ability of the steam generator blowdown system to meet the applicable acceptance criteria. Therefore, the staff finds that the proposed changes to AP1000 DCD, Section 10.4.8, are acceptable.

10.4.10 Auxiliary Steam System

The staff approved Section 10.4.10, "Auxiliary Steam System," of the AP1000 DCD, Revision 15, in the certified design. In the the proposed changes to the AP1000 DCD, the applicant proposed the following change to Section 10.4.10 of the certified design.

The applicant replaced the oil-fired boiler with an electric boiler. TR-114, "AP1000 Auxiliary Boiler Sizing and Design" (APP-GW-GLR-114), dated June 14, 2007, documents the basis for this change. The applicant deleted detailed design information for the oil-fired boiler and proposed some changes throughout Section 10.4.10 to reflect the changes addressed in TR-114.

The staff confirmed that all other changes in DCD Section 10.4.10 are editorial and do not require an evaluation.

10.4.10.1 Evaluation

The staff reviewed all changes identified in the AP1000 DCD. The staff did not review descriptions and evaluations of the auxiliary steam system in the AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes. All technical changes in the DCD are supported by information presented in Westinghouse TRs. The specific criteria that apply to the auxiliary steam system related changes include 10 CFR 52.63(a)(1)(vi) in that the changes substantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security, and 10 CFR 52.63(a)(1)(vii), in that the proposed changes contribute to increased standardization of the certification information.

NUREG-1793 includes the regulatory basis for Section 10.4.10 of the AP1000 DCD. In its previous evaluations of this section, the staff found that the current NUREG-0800 does not include a section specifically addressing the auxiliary steam system. The staff has determined that the acceptability of this system will be based on meeting the requirements of GDC 4. In other words, failure of the auxiliary steam system, as a result of a pipe break or malfunction of the system, should not adversely affect safety-related SSCs. The staff has reviewed the proposed changes to DCD Section 10.4.10 and discusses the results below.

The applicant replaced the oil-fired boiler with an electric boiler and submitted TR-114 to address the basis for this change. The applicant proposed a series of changes to DCD Section 10.4.10 to be consistent with the new electric boiler. These changes include the following:

- The applicant added text to DCD Section 10.4.10.1.2 to indicate that auxiliary steam supplements the main steam system during startup. This change provides additional information on system operation, has no impact on the auxiliary steam system and is, therefore, acceptable.

- The applicant deleted reference to “oil-fired boiler” and replaced it with “electric package boiler” in DCD Section 10.4.10.2.2. This is a conforming change to reflect the design change and is, therefore, acceptable.

- The applicant changed the nominal net output capacity from “at least 110,000 pounds per hour (49,900 kilograms per hour)” to “approximately 100,000 pounds per hour (45,360 kilograms per hour)” in DCD Section 10.4.10.2.2. This change has no impact on any safety-related SSC and, therefore, is acceptable.

- The applicant deleted the component description for the auxiliary boiler fuel-oil components from DCD Section 10.4.10.2.2. This is a conforming change to reflect the design change and, therefore, is acceptable.

10.4.10.2 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the application for DC met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

The staff reviewed the applicant's proposed changes to the AP1000 auxiliary steam system against GDC 4. The staff finds that the applicant's proposed changes do not affect the ability of the auxiliary steam system to meet the applicable acceptance criteria. The staff also finds that the applicant properly incorporated the design changes into the appropriate sections of the AP1000 DCD. The staff concludes that the AP1000 auxiliary steam system continues to meet all applicable acceptance criteria, and the proposed changes are properly documented in the updated AP1000 DCD; therefore, the staff finds that the proposed changes to AP1000 DCD, Section 10.4.10, are acceptable.

11. RADIOACTIVE WASTE MANAGEMENT

The AP1000 radioactive waste (radwaste) management systems control the handling and treatment of liquid, gaseous, and solid radwaste. These systems include the liquid radwaste system (WLS), the gaseous radwaste system (WGS), and the solid radwaste system (WSS). The WLS is designed to control, collect, process, store, and dispose of liquid radioactive wastes. The WLS is discussed in Section 11.2 of this report. The WLS contains holdup tanks, process pumps, and other processing equipment, including monitor tanks and appropriate instrumentation and controls. Ion exchange is the principal waste treatment process in the WLS.

The WGS collects, processes, and monitors gaseous releases. The WGS is discussed in Section 11.3 of this report. The WGS collects gaseous wastes that are potentially radioactive or hydrogen-bearing (i.e., those wastes resulting from degassing the reactor coolant and the contents of the reactor coolant drain tank (RCDT)), stores them for decay in charcoal delay beds, and subsequently releases them into the environment via the plant vent. The WSS controls the processing of solid wastes generated during reactor operation, as well as the packaging and storage of such processed wastes before shipment to a licensed disposal facility. The WSS is discussed in Section 11.4 of this report. The process and effluent radiological monitoring instrumentation and sampling systems, which are discussed in Section 11.5 of this report, detect and measure the radioactive materials in plant liquid and gaseous processes and effluent streams.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the additional and amended information provided by the applicant using the guidance in Chapter 11 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Revision 3, issued March 2007. The NRC developed the original NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” using the guidance from regulatory guide (RG) 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” Revision 3, issued November 1978. Therefore, this supplement follows the format of the original NUREG-1793.

The scope of this review includes Chapter 11, Sections 11.2 through 11.5 of Revision 16 and 17 of the AP1000 design certification amendment, Tier 2 and Tier 1, Chapter 2, Sections 2.3.10, “Liquid Radwaste System”; 2.3.11, “Gaseous Radwaste System”; and 3.5, “Radwaste Monitoring,” which includes the associated inspections, tests, analyses, and acceptance criteria (ITAAC). Section 11.1 did not include any technical changes in Revisions 16 and 17; therefore, the staff did not include Section 11.1 in this supplement.

This section describes the staff’s evaluation and findings of the AP1000 Design Control, Document (DCD), Revisions 16 and 17. The staff reviewed Revision 16 upon receipt; however, after review, the staff issued several requests for additional information (RAIs) and the applicant issued Revision 17 prior to the resolution of these RAIs. Hence, this review and evaluation encompasses both revisions.

11.2 Liquid Waste Management System

11.2.1 Summary of Technical Information

In the AP1000 DCD, the applicant proposed to make five technical changes as described below:

1) In AP1000 DCD Section 11.2, the applicant proposed to increase overall liquid waste holdup capacity and improve operational flexibility by adding three additional liquid waste monitor tanks (and associated pumps, piping, instruments, and valves). The applicant proposed to house the new tanks 56,781 liters (15,000 gallons) each in the radwaste buildings. The new tanks are identical to the three existing monitor tanks, which are housed in the auxiliary building. The applicant has documented these changes in Westinghouse TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension," APP-GW-GLN-116, Revision 0.

2) In AP1000 DCD Sections 11.2.1.2.4 and 11.2.1.3, the applicant added statements certifying compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1406, "Minimization of contamination." These changes are documented in Westinghouse TR-98, "Compliance with 10 CFR 20.1406," APP-GW-GLN-098, Revision 0.

3) In AP1000 DCD Section 11.2, Figure 11.2-2, "Liquid Radwaste System Piping and Instrumentation Diagram, Sheet 1, the applicant proposed changing the figure to show the addition of a flow element to monitor letdown flow from the chemical and volume control system to the liquid waste system. The applicant documented this chemical and volume control system post accident monitoring system (PAMS) instrument modification in TR-118, "Chemical and Volume Control System PAMS Instrument Modifications," APP-GW-GLN-118, Revision 0.

4) In AP1000 DCD Sections 11.2.5.3 and 11.2.5.4, the applicant proposed updates to document the closure of combined license (COL) Information Items 11.2-3 and 11.2-4, respectively and as described in TR-48, "Identification of Ion Exchange and Adsorbent Media," APP-GW-GLR-008, Revision 0 and in TR-73, "Dilution and Control of Boric Acid Discharge," APP-GW-GLR-014, Revision 0.

5) In AP1000 DCD Section 11.2.3.3, 11.2.3.5 and 11.2.5.2, the applicant proposed changes related to the liquid effluent release requirements in 10 CFR Part 20, "Standards for protection against radiation," and 10 CFR Part 50, "Domestic licensing of production and utilization facilities."

The changes in Revision 17 to the AP1000 DCD reflect the responses to the RAIs from Revision 16.

The evaluation below discusses these changes and other information pertaining to NUREG-0800 acceptance criteria, provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.2.2 Evaluation

The staff reviewed all technical changes and ITAAC to the liquid waste management system in accordance with NUREG-0800 Section 11.2, "Liquid Waste Management System" and all changes identified by change marks in the AP1000 DCD. In addition, the staff reviewed all changes to the liquid waste management system-based design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.10.

11.2.2.1 Addition of Three Liquid Waste Monitor Tanks

In TR-116, the applicant proposed to add three additional liquid waste monitor tanks (and associated pumps, piping, instruments, and valves). The applicant proposed to house the new tanks, 56,781 liters (15,000 gallons) each in the radwaste building.

In AP1000 DCD Tier 2, Section 11.2, the staff noted that a potential exists for the quantity of radionuclides in the radwaste building portion of the liquid waste management system to exceed the A_1 value, thus requiring these structures, systems, and components (SSCs) to be designated as RW-IIa in accordance with RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Revision 2. However, these SSCs were designated as RW-IIc in AP1000 DCD Tier 2, Section 1, Appendix 1A. In a September 9, 2008 letter, the applicant responded to the staff’s concern and provided an analysis to show that the concentrations will be less than the A_1 and A_2 levels. The applicant then modified Revision 17 of the AP1000 DCD to state that the contents of each monitor tank in the non-seismic radwaste building will be less that the A_1 and A_2 levels of 10 CFR Part 71, “Packaging and transportation of radioactive material,” Appendix A, “Determination of A_1 and A_2,” Table A-1, “A_1 and A_2 Values for Radionuclides.” The staff finds this response acceptable and RAI-SRP11.2-CHPB-06 is closed.

In AP1000 DCD Section 3.2, Table 3.2-3, Sheets 63 and 64, the staff noted that the additional equipment for the WLS was not included in the table, nor was the location of the added WLS components included in the Radwaste Building. Moreover, in Table 3.2-3, the applicant did not identify the new WLS piping interconnecting the Auxiliary and Radwaste Buildings and its classification. The applicant added the necessary information to the diagrams and table in Revision 17. The staff reviewed these changes and found them acceptable. RAI-SRP11.2-CHPB-07 is closed.

Based on the evaluation of AP1000 DCD Section 11.2 and the response to the RAI, the staff concludes that the applicant properly identified all design information related to the three additional liquid waste monitor tanks and associated equipment, and provided an adequate demonstration that design objectives for equipment necessary to control releases of radioactive effluents to the environment have been met in accordance with 10 CFR Part 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.”

11.2.2.2 Documentation of Compliance with 10 CFR 20.1406

In TR-98, the applicant proposed to comply with the regulation by selection of design technology. Table TR98-1, “AP1000 Features Applicable to 10 CFR 20.1406” in TR-98 lists specific examples of how the AP1000 design complies with 10 CFR 20.1406 (Items 19, 22, 23, 24, 25, and 26). The staff reviewed the items listed in Table TR98-1 pertaining to the WLS and found that the applicant addressed the minimization of waste generation in 10 CFR 20.1406.

The staff issued regulatory guidance for 10 CFR 20.1406 in RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” after Revision 16 but before the issuance of Revision 17. As such, the staff documented its review according to

RG 4.21 in Chapter 12. The staff finds that this change does not affect the design and performance aspects of the WLS.

11.2.2.3 Chemical and Volume Control System Post Accident Monitoring System Instrument Modifications

In AP1000 DCD Section 11.2, Figure 11.2-2, “Liquid Radwaste System Piping and Instrumentation Diagram” Sheet 1, the applicant changed the figure to include a flow element to monitor letdown flow from the chemical and volume control system to the WLS. The staff concluded that this change does not affect the design and performance aspects of the WLS and provides additional capabilities to monitor the flow to the WLS.

11.2.2.4 Closure of COL Information Item 11.2-3 and 11.2-4

In AP1000 DCD Sections 11.2.5.3 and 11.2.5.4, the applicant proposed updates to document the closure of COL Information Items 11.2-3 and 11.2-4, respectively as described in TR-48 and in TR-73. The applicant proposed that media selection should be a matter for the plant operator, and should not be identified at the COL application stage. The two reports, TR-48 and TR-73, describe this process and give justification to show that COL Information Items 11.2-3 and 11.2-4 should be deleted. For both of these applications, the staff considered that this media will be replaced multiple times throughout the operating life of the plant. Since these changes are operational in nature and do not affect the design and performance aspects of the WLS, the staff finds these changes acceptable.

11.2.2.5 Changes Related to the Effluent Release Requirements in Parts 20 and 50

Section 11.2.3.3, “Dilution Factor” in the AP1000 DCD, Revision 16 omitted compliance with the annual offsite dose limits of 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” Section II.A. The DCD stated that the plant operator selected dilution flow rates to ensure compliance with the effluent concentration limits of 10 CFR Part 20 and any local requirements. Plant operators must also ensure that the annual releases are within the dose limits of 10 CFR Part 50, Appendix I, which in some situations may be the limiting case. Because of this omission, the staff issued an RAI requesting the applicant to incorporate Section II.A requirements. The applicant added this requirement to Revision 17. The staff reviewed this change and concluded that it was acceptable and closed RAI-SRP11.2-CHPB-01.

Section 11.2.3.5 in the AP1000 DCD, Revision 16 stated that the estimated doses from liquid effluents are site-specific and discussed in DCD Section 11.2.5. Section 11.2.5 only stated that the COL applicant will provide a site-specific cost-benefit analysis to address the requirements of 10 CFR Part 50, Appendix I, Section II.D regarding population doses due to liquid effluents. The COL applicant must also comply with the individual dose limits to members of the public in 10 CFR Part 50, Appendix I and 10 CFR 20.1301(e). Because of this omission, the staff issued an RAI requesting the applicant to incorporate the individual dose requirements of 10 CFR Part 20 and 10 CFR Part 50. The applicant added these requirements to Revision 17. The staff reviewed this change and concluded that it was acceptable and closed RAI-SRP11.2-CHPB-02.

The staff found an inconsistency between the estimated dose sections in Sections 11.2, “Liquid Waste Management System,” and 11.3, “Gaseous Waste Management System.” No individual doses were calculated for liquid effluents, but doses were calculated for gaseous releases. The staff issued an RAI asking the applicant to address this inconsistency. Revision 17 to the AP1000 DCD addressed this inconsistency by adding text that requires compliance with Section II.A of Appendix I to 10 CFR Part 50. The staff reviewed this change and concluded that it was acceptable and closed RAI-SRP11.2-CHPB-03.

11.2.2.6 Preoperational Testing Information

Section 11.2.4, “Preoperational Testing,” of the AP1000 DCD, Revisions 16 and 17 did not address the testing and inspection of ion exchange resin. The initial performance of the liquid radioactive waste system depends on the existence and performance of ion exchange resin in the ion exchange vessels. The applicant based the annual liquid effluent release of radioactivity estimated in Section 11.2 on assuming that the media provided a specific level of decontamination as listed in Table 11.2-5, “Decontamination Factors.” The applicant did not specify any preoperational testing and inspections to ensure that the resin is properly installed and performing to assumed levels at initial start up.

The staff issued RAI-SRP11.2-CHPB-05 requesting a statement in the DCD indicating that the applicant will confirm the presence of the correct amount of resin in the liquid radwaste system ion exchange vessels. On April 3, 2009, the applicant submitted a proposed revision to DCD Section 11.2.4.3. This revision stated that an inspection of the system would confirm that the applicant installed the proper volume of media into the appropriate components. The staff finds this confirmation acceptable. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

11.2.2.7 Reactor Coolant System Effluents

In the AP1000 DCD, Revision 17, the applicant included additional operator actions when returning the contents of the monitoring tanks for further processing. When radioactivity exceeds operational targets, the operator is instructed to recirculate and sample the contents of the monitoring tanks. The staff reviewed this change and found it to be more specific and appropriate. Thus, the staff concluded that this change was acceptable.

11.2.2.8 Tier 1 Section 2.3.10, Liquid Radwaste System

The applicant added three additional WLS monitoring tanks and a letdown flow monitor to reflect the current design changes in Tier 1, Section 2.3.10. These are conforming changes and have no impact on the staff’s conclusion about the acceptability of the design of the WLS.

11.2.3 Conclusion

In NUREG-1793, Supplement 1, the staff documented its conclusion that the AP1000 design and DCD (up to and including Revision 15 of the DCD) were acceptable and that the applicant’s application for the design certification met the requirements of Subpart B to 10 CFR Part 52, “License, certifications, and approvals for nuclear power plants” that are applicable and technically relevant to the AP1000 standard plant design.

In the previous evaluation of AP1000 DCD, Section 11.2, “Liquid Waste Management System,” the staff identified acceptance criteria based on the design’s meeting relevant requirements in

10 CFR 20.1302, “Compliance with dose limits for individual members of the public,” as it relates to limits on doses to persons in unrestricted areas; 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment; 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants” General Design Criteria (GDC) 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to the design of liquid waste management systems to control releases of liquid radioactive effluents; and GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to the design of liquid waste management systems to ensure adequate safety under normal and postulated accident conditions.

The staff reviewed the applicant’s proposed changes to the AP1000 liquid waste management system. The staff concluded that the applicant’s proposed changes do not affect the ability of the AP1000 liquid waste management system to meet the applicable acceptance criteria in NUREG-0800 Section 11.2. The staff also concluded that the applicant had properly incorporated design changes into the appropriate sections of the AP1000 DCD. The staff determined that the AP1000 liquid waste management system continued to meet all applicable acceptance criteria and the applicant documented the changes in the updated AP1000 DCD. The staff also concluded that all of the changes related to the liquid waste management system design description and ITAAC as described in AP1000 DCD Tier 1, Section 2.3.10 are acceptable.

11.3 Gaseous Waste Management System

11.3.1 Summary of Technical Information

In the AP1000 DCD, Revision 17, the applicant proposed ten technical changes to the gaseous waste management system as supported by information presented in TR-48, TR-98, and TR-103, “Fluid System Changes” APP-GW-GLN-019, Revision 2. A description of each Revision 16 technical change is provided below:

1) The applicant demonstrated compliance with 10 CFR 20.1406 through the selection of design technology. This change is documented in TR-98. The applicant identified this change in AP1000 DCD, Revision 16, Tier 2, page 11.3-4.

2) The applicant completed COL Information Item 11.3.5.2 by providing the requested information in TR-48. The applicant has identified this change in AP1000 DCD, Revision 16, Tier 2, pages 11.3-4 and 11.3-11.

3) The applicant changed Figure 11.3-1, “Gaseous Radwaste System Piping and Instrumentation Diagram,” to update the drawing nomenclature. This change was documented TR-103. The applicant identified this change in AP1000 DCD, Revision 16, Tier 2, page 11.3-20.

4) On page 11.3-2 in the DCD section on water incursion (Section 11.3.1.2.2.2), the applicant added the automatic isolation of the guard bed inlet on high moisture separator level.

5) In the General Description (Section 11.3.2.1) on page 11.3-5, the applicant changed the temperature of the influent gas from 7 degrees Celsius (C) (45 degrees Fahrenheit (F)) to 4 degrees C (40 degrees F).

6) In the General Description (Section 11.3.2.1) on page 11.3-5 and in Section 11.3.2.3.3 on pages 11.3-7 and 11.3-8, the applicant reduced the capacity of each of the two activated carbon delay beds from 100 percent system capacity to 50 percent capacity.

7) In order to maintain a slight positive pressure in the WGS when the system is inactive, the applicant originally proposed to inject a small nitrogen flow into the system. In Revision 17, the applicant eliminated the nitrogen injection and relied on a closed discharge isolation valve to maintain positive pressure. The applicant documented this change in Section 11.3.2.2.1, "Normal Operation," on page 11.3-5.

8) The applicant previously stated that the gas leaving the moisture separator would be monitored for moisture content and a high alarm would alert the operator to a condition requiring attention. The applicant revised the design and operation of the system to monitor temperature and not moisture. The applicant documented this change in Section 11.3.2.2.1, "Normal Operation," on page 11.3-6 and in Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System."

9) The applicant removed the Xenon and Krypton dynamic adsorption coefficients and holdup times from Table 11.3-1, "Gaseous Radwaste System Parameters," on page 11.3-12.

10) The applicant removed some of the parameter data and revised some of the remaining parameter values in Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System" on page 11.3-13.

The evaluation below discusses these changes and other information pertaining to the NUREG-0800 acceptance criteria, provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.3.2 Evaluation

The staff reviewed all technical changes to the WGS in accordance with NUREG-0800 Section 11.3, "Gaseous Waste Management System" and all changes identified by change marks in the AP1000 DCD. In addition, the staff reviewed all changes to the gaseous waste management system design description and ITAAC in AP1000 DCD, Tier 1, Section 2.3.11.

11.3.2.1 Compliance with 20.1406

In TR-98, the applicant proposed to comply with the regulation by the selection of design technology. Table TR-98-1 in TR-98 lists specific examples of how the AP1000 design complies with 10 CFR 20.1406 (Items 19, 22, 23, 24, 25, and 26). The staff reviewed the items listed in Table TR-98-1 pertaining to the WGS and found that the applicant addressed the minimization of waste generation in 10 CFR 20.1406.

The NRC staff issued regulatory guidance for 10 CFR 20.1406 in RG 4.21 after Revision 16 but before the issuance of Revision 17. As such, the staff documented its review according to RG 4.21 in Chapter 12. The staff finds that this change does not affect the design and performance aspects of the gaseous waste management system.

11.3.2.2 Completion of COL Item 11.3.5.2

The applicant addressed COL Information Item 11.3.5.2 by providing the requested information in TR-48. The applicant proposed that media selection should be a matter for the plant operator, and should not be identified at the COL application stage. The two reports, TR-48 and TR-73, describe this process and justify the deletion of COL Information Item 11.3.5.2. For both of these applications, the staff considered that this media was a consumable item, designed for replacement when expended. The media will be replaced multiple times throughout the operating life of the plant. For these reasons, the staff concluded that these changes are operational in nature and do not affect the design and performance aspects of the WGS.

11.3.2.3 Revision of Figure 11.3-1 Piping and Instrumentation Diagram

The applicant proposed a change to Figure 11.3-1, “Gaseous Radwaste System Piping and Instrumentation Diagram,” to update the drawing nomenclature. The staff concluded that the updated nomenclature did not affect the design or performance of the system and considered the change in nomenclature editorial.

11.3.2.4 Addition of Automatic Isolation of the Guard Bed Inlet on High Moisture Separator Level

In Section 11.3.1.2.2.2, “Water Incursion,” the applicant removed an automatic isolation of the guard bed inlet on high moisture separator level. The staff concluded that this change was acceptable because the system has an automatic isolation on temperature, a moisture separator, drain traps, and the guard bed. These all provide protection from wetting of the activated carbon delay beds. Activated carbon loses its retention properties when wet.

11.3.2.5 Temperature of the Influent Gas Changed From 7 °C (45 °F) to 4 °C (40 °F)

In Section 11.3.2.1, “General Description,” the applicant changed the outlet gas temperature from the gas cooler from about 7 °C (45 °F) to about 4 °C (40 °F). This change is small and conservative, and does not affect the conclusions in NUREG-1793. The staff found this change acceptable.

11.3.2.6 Reduced Capacity of Each of the Two Activated Carbon Delay Beds from 100 Percent System Capacity to 50 Percent Capacity

In Section 11.3.2.3.3, “Gaseous Radwaste System Tanks,” the applicant proposed to reduce each activated carbon delay bed capacity from 100 percent of the system capacity to 50 percent of the system capacity during design basis conditions. A single bed still provided adequate performance under normal conditions. The staff issued RAI-SRP11.3-CHPB-04 requesting additional information to verify compliance with NUREG-0800 Section 11.3, “SRP Acceptance Criteria,” Item 2 regarding the capacity of the system to meet the anticipated processing requirements of the plant.

On April 3, 2009, the applicant responded to RAI-SRP11.3-CHPB-04 and provided an analysis to determine the effects of operation with one charcoal delay bed out of service. The applicant used the PWR-GALE code to determine gaseous effluent releases assuming delay times for noble gases with only one delay bed in service. The applicant then compared the expected airborne concentrations and doses at the site boundary to those using the holdup times for two delay beds. The results of the analysis for the one delay bed showed that the effluent concentrations and doses at the site boundary increased by negligible amounts and were still well within the limits of 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” Table 2, and the design limits of Appendix I to 10 CFR Part 50.

The staff reviewed this analysis and concluded that the applicant used reasonable delay times and an approved computer code to simulate the performance of one delay bed. The staff found the response acceptable and RAI-SRP11.3-CHPB-04 is closed.

11.3.2.7 Elimination of the Nitrogen Injection and Reliance on a Closed Discharge Isolation Valve to Maintain Positive Pressure in WGS

In Section 11.3.2.2.1, the applicant used the discharge isolation valve to maintain the WGS at a positive pressure. In AP1000 DCD Section 11.3.2.3.4, “Remotely Operated Valves,” the applicant provided information regarding the nitrogen purge pressure control valve. The applicant stated that this valve maintains a small positive pressure in the WGS to prevent ingress of air during periods of low flow. Section 11.3.1.2.3.1, “Prevention of Hydrogen Ignition,” stated that the discharge isolation valve of the WGS is continuously pressurized with nitrogen to prevent ingress of air into the system from the discharge path. The staff concluded that this change is acceptable as a provision to preclude the ingress of air and does not affect the conclusions in NUREG-1793.

11.3.2.8 Monitoring Temperature Instead of Moisture of the Gas Leaving the Moisture Separator

In Section 11.3.2.2.1, “Normal Operation,” and Table 11.3-2, “Component Data (Nominal) – Gaseous Radwaste System of Instrument Indication and Alarms,” the applicant changed the monitored parameter of the gas leaving the moisture separator to temperature instead of moisture. The high temperature alarm indicates a reduced performance of the moisture separators and alerts the operator of an abnormal situation. Since this change performs the same function as the moisture alarm in the previous revisions, the staff finds this change acceptable.

11.3.2.9 Removal of the Xenon and Krypton Dynamic Adsorption Coefficients and Holdup Times from Table 11.3-1

In AP1000 DCD Table 11.3-1, “Gaseous Radwaste System Parameters,” the applicant removed the dynamic adsorption coefficients and holdup times for noble gases. With the removal of this information, the staff questioned the justification of the holdup times used to calculate the Krypton and Xenon releases in the gaseous effluents. In addition, the staff used these holdup times to conclude that no alteration in system operation would be necessary due to adverse meteorological conditions. The staff asked the applicant in RAI-SRP11.3-CHPB-04 to explain the reason for removing the coefficients and holdup times from the table.

On April 3, 2009, the applicant responded to the staff’s concern and stated that the dynamic adsorption coefficients and holdup times were removed from the table to avoid confusion. The applicant did not use these data in any analysis. Furthermore, these coefficients and holdup times are less conservative than what the applicant used to calculate the release of noble gases using the GALE code.

The staff confirmed that the applicant did not use these data in any analysis; therefore, any conclusions regarding effluent releases remain unchanged. Even using the more conservative dynamic coefficients and holdup times from the GALE code, the resulting concentrations and doses are low enough to provide an ample margin of safety should adverse meteorological conditions occur. The staff found the response acceptable and closed RAI-SRP11.3-CHPB-04.

11.3.2.10 Removal of Some of the Parameter Data and Revising Some of the Remaining Parameter Values in Table 11.3-2

The applicant revised AP1000 DCD Table 11.3-2, "Component Data (Nominal) – Gaseous Radwaste System" to reflect the performance of the current design. The changes include pump operating pressure, heat exchanger type, and heat exchanger operating temperature, pressure, and flow. These changes have been reviewed by the staff and have been found to be minor, do not affect the performance of the system to meet the design objectives and the conclusions in NUREG-1793, and are, therefore, acceptable.

11.3.2.11 Additional NUREG-0800 Section 11.3 Acceptance Criteria

The NRC staff noted that Section 11.3.3 was missing the consequence evaluation of a gaseous waste system leak or failure. The acceptance criteria in NUREG-0800 Section 11.3 are based on the availability of this information as part of the evaluation and relies on the approach specified in Branch Technical Position (BTP) 11-5. Based on the safety evaluation report (SER) for Revision 15, the applicant performed this analysis in response to an RAI, but the description of the analysis and results were not included in the AP1000 DCD. The staff issued RAI-SRP11.3-CHPB-02 requesting the inclusion of this analysis in the DCD.

On January 13, 2009, the applicant responded to RAI-SRP11.3-CHPB-02 stating that Section 11.2.2 will be revised to include the consequence analysis. The applicant committed to base its analysis on 1 percent fuel defects, 1-hour bypass of charcoal beds, 30 minute decay prior to release to the environment, and updated atmospheric dispersion factors. At that time, the applicant had not submitted the consequence analysis. This issue was tracked as an open item in the SER until the staff received and evaluated the resulting analysis.

On July 9, 2009, the applicant responded to RAI-SRP11.3-CHPB-02 Revision 1, with the requested analysis and proposed revisions to the AP1000 DCD. The applicant based its analysis on BTP 11-5, which included 1 percent fuel defects, 1-hour bypass of the charcoal beds, 30 minute decay prior to release to the environment, and updated atmospheric dispersion factors. The staff reviewed this response and determined that the applicant properly followed BTP 11-5. The staff independently verified the calculations. Based on the facts that the applicant's analysis properly followed BTP 11-5, the staff verified the results, and the results meet the acceptance criteria in BTP 11-5, the staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

11.3.2.12 Tier 1 Section 2.3.11, Gaseous Radwaste System

The applicant revised this section to indicate that the activated carbon delay beds will be designed to one-half seismic Category I instead of full seismic Category I. Since the WGS is not a safety-related system, the staff finds that this change is in accordance with RG 1.143 and is acceptable.

Additionally, in a letter dated January 20, 2010, the applicant also proposed Change Number 23 to AP1000 DCD, Tier 1 Section 2.3.11 removing all ITAAC concerning the seismic design loads of the equipment in the gaseous radioactive waste management system, specifically the carbon delay beds and the discharge isolation valve. In RAI-SRP11.3-CHPB-06, the staff questioned the basis for removing the ITAAC because these components have seismic design criteria that the applicant must inspect and test before operation.

In response to this RAI, the applicant reinstated the ITAAC for the gaseous waste management system and provided seismic design criteria. The staff reviewed this response and found that the ITAAC meet the inspection and testing recommendations in RG 1.143 and finds the response acceptable.

In a subsequent revision to the DCD, the applicant revised DCD Tier 1, Section 2.3.11, and Table 2.3.11-1 to include the appropriate ITAAC, which resolves this issue.

11.3.3 Conclusion

In NUREG-1793, Supplement 1, the staff concluded that the AP1000 design and DCD (up to and including Revision 15 of the DCD) was acceptable and that the application for the design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In the previous evaluation of the AP1000 DCD, Section 11.3, “Gaseous Waste Management System,” the staff identified acceptance criteria based on the design’s meeting relevant requirements in 10 CFR 20.1302, as it relates to limits on doses to persons in unrestricted areas; 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment; GDC 60, as it relates to the design of the gaseous waste management system to control releases of radioactive effluents; and GDC 61 as it relates to the design of the gaseous waste management systems to ensure adequate safety under normal and postulated accident conditions.

The staff reviewed the applicant’s proposed changes to the AP1000 gaseous waste management system as documented in the AP1000 DCD. The staff concluded that the applicant’s proposed changes do not affect the ability of the AP1000 gaseous waste management system to meet the applicable acceptance criteria in NUREG-0800 Section 11.3. The staff also concluded that the applicant had properly incorporated design changes into the appropriate sections of AP1000 DCD. The staff determined that the AP1000 gaseous waste management system continued to meet all applicable acceptance criteria and the applicant documented these changes in the updated AP1000 DCD. The staff also concluded that all of the changes related to the gaseous waste management system design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 are acceptable.

11.4 Solid Waste Management System

11.4.1 Summary of Technical Information

In the AP1000 DCD, Revision 16, the applicant proposed to make changes to Section 11.4. The applicant proposed two technical changes that were supported by information presented in TR-98 and TR-103. The description of each technical change is provided below:

1) The applicant proposed to demonstrate compliance with 10 CFR 20.1406 through the selection of design technology. This change is documented in TR-98. The applicant has identified this change in the AP1000 DCD, Revision 16, Tier 2, on pages 11.4-3 and 11.4-14.

2) The applicant proposed to change the type of pump used for spent resin transfer. This change is documented in TR-103. The applicant has identified this change in the AP1000 DCD, Revision 16, Tier 2, Table 11.4-10, and on page 11.4-32.

The applicant made no additional technical changes in subsequent revisions to the AP1000 DCD. The evaluation below discusses technical changes provided in updated revisions to the AP1000 DCD, and provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.4.2 Evaluation

The staff reviewed all changes to the solid waste management system as described in the AP1000 DCD in accordance with the guidance in NUREG-0800 Section 11.4, "Solid Waste Management System." The regulatory basis for Section 11.4 of the AP1000 DCD is documented in NUREG-1793. The following evaluation discusses the results of the staff's review. In addition, the staff reviewed the solid waste management system design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 and identified no changes to this section.

11.4.2.1 Documentation of Compliance with 10 CFR 20.1406

In TR-98, the applicant proposed to comply with the regulation by the selection of design technology. Table TR98-1 in TR-98 lists specific examples of how the AP1000 design complies with 10 CFR 20.1406 (Items 19, 22, 23, 24, 25, and 26). The staff reviewed the items listed in Table TR98-1 pertaining to the solid waste management system and found that the applicant addressed the minimization of waste generation in 10 CFR 20.1406.

The NRC staff issued regulatory guidance for 10 CFR 20.1406 in RG 4.21 after the receipt of DCD Revision 16, but before the issuance of Revision 17. As such, the staff documented its review according to RG 4.21 in Chapter 12. The staff concluded that this change does not affect the design and performance aspects of the solid waste management system.

11.4.2.2 Spent Resin Transfer Pump

In TR-103, the applicant stated that utilities have reported operational problems due to wear of progressive cavity pumps, and the grinding of resin beads. The crushed beads are difficult to dewater and could create a storage problem. To remedy this situation, the applicant proposed to replace the progressive cavity pump with a material handling positive displacement pump. The staff concluded that this change does not affect the design of the WSS. The staff finds that the conclusions of NUREG-1793 regarding the acceptability of the WSS remain valid.

11.4.3 Conclusion

In NUREG-1793, Supplement 1, the staff concluded that the AP1000 design was acceptable and that the applicant's application for the design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In the previous evaluation of AP1000 DCD, Section 11.4, "Solid Waste Management System," the staff identified acceptance criteria based on the design's meeting the relevant requirements in 10 CFR 20.1302, as it relates to limits on dose to persons in unrestricted areas; 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate the design objectives for equipment necessary to control releases of radioactive effluents to the environment; GDC 60, as it relates to the design of waste management systems to control releases of radioactive effluents; and GDC 61, as it relates to the design of waste management systems to ensure adequate safety under normal and postulated accident conditions.

The staff reviewed the applicant's proposed changes to the AP1000 solid waste management system as documented in the the latest revision of the AP1000 DCD. The staff concluded that the applicant's proposed changes do not affect the ability of the AP1000 solid waste management system to meet the applicable acceptance criteria in NUREG-0800 Section 11.4. The staff also concluded that the applicant had properly incorporated design changes into the appropriate sections of the AP1000 DCD. The staff determined that the AP1000 solid waste management system continued to meet all applicable acceptance criteria and the applicant documented the changes in the updated AP1000 DCD. The staff also concluded that all of the changes related to the gaseous waste management system design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 were acceptable. In addition, the staff reviewed the solid waste management design description and ITAAC in AP1000 DCD Tier 1, Section 2.3.11 and found no changes to this section.

11.5 Radiation Monitoring

This section describes the staff's evaluation and findings of the AP1000 DCD, Revisions 16 and 17. The staff reviewed Revision 16 upon receipt and after review issued several RAIs. Prior to resolving these RAIs, the applicant issued Revision 17 to the DCD. Hence, this review and evaluation encompasses both revisions.

11.5.1 Summary of Technical Information

In the AP1000 DCD, Revision 16, the applicant made changes to Section 11.5. In Revision 16, the applicant only made editorial changes by renaming the Technical Support Center to the Control Support Area. The staff reviewed these changes and found them to be editorial in nature.

In Revision 17, the applicant proposed the following five technical changes:

1) The applicant switched the radiation monitor for the service water blowdown from an offline monitor to an inline monitor. The applicant documented this change on pages 11.5-5 in Section 11.5.2.3.1.

2) The applicant eliminated the statement that it will follow the design guidelines of American National Standards Institute (ANSI) N13.1, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities" for the discharge radiation monitor of the turbine island vent. The applicant documented this change in Section 11.5.2.3.3 on page 11.5-11.

3) The applicant switched the radiation monitor for the liquid radwaste discharge from an offline monitor to an inline monitor. The applicant documented this change on pages 11.5-11 and 11.5-12 in Section 11.5.2.3.3.

4) The applicant made several changes to the monitors and specifications listed in Table 11.5-1. These changes included the following:

   - addition of two main steam line monitors for N-16
   - lowering the minimum and maximum of the nominal range of the steam generator blowdown monitors by an order of magnitude
   - adding the detection of Ar-51 and N-13 to the containment atmosphere monitors
   - lowering the minimum and maximum of the nominal range for the fuel handling building, auxiliary building, and annex building exhaust vents by an order of magnitude
   - changing the nominal range for the monitors of the main control room air supply duct, containment air filtration exhaust, health physics (HP) and hot machine shop exhaust, radwaste building exhaust, and the gaseous and liquid radwaste discharge

5) The applicant made several changes to the monitors and specifications listed in Table 11.5-2. These changes included the following:

   - the addition of a liquid and gaseous monitor for radwaste area 2
   - the addition of a containment area personal hatch monitor at maintenance elevation level 30.48 meters (m) (100.0 feet (ft))
   - specifying the location of the containment area monitor at operating deck – 41.22 m (135 ft, 3 inches (in))
   - removing the rail car bay area as a service area for monitor RMS-JE-RE013

The evaluation below discusses these changes and other information, and provides an overview of the staff's RAIs, the applicant's responses, and the staff's evaluation of the responses.

11.5.2 Evaluation

The staff reviewed the entire section and all technical changes to the radiation monitoring systems in accordance with NUREG-0800 Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems." The staff reviewed the entire section and all changes identified by change marks in the AP1000 DCD.

11.5.2.1 Offline to Inline Monitors for Service Water and Radwaste Liquid Discharges

The change from offline to inline monitoring for the service water blowdown, liquid radwaste discharge, and the waste water discharge has the advantage of reducing the potential for liquid leaks, reducing the likelihood of equipment failure, and reducing areas within the system where radioactivity can accumulate and present exposure problems. For these reasons, the staff concluded that the change from offline to inline monitoring is acceptable.

11.5.2.2 Removal of Commitment to ANSI N13.1 for Turbine Vent Monitor

The applicant removed the commitment to follow ANSI N13.1 for the design of the monitoring system for the turbine island vent. This action raises a question about what design guidance the applicant would follow for this vent. NUREG-0800 Section 11.3 states that ANSI/HPS N13.1-1999 should be used. The staff issued RAI-SRP11.5-CHPB-02 and RAI-SRP11.5-CHPB-03 requesting that the applicant specify what design guidance it will follow.

On January 13, 2009 and March 23, 2009, the applicant responded to the RAIs by stating that since the turbine island vent monitor is an inline noble gas monitor and not a sample extraction monitor, ANSI N13.1 does not apply. The applicant then further stated that the monitor complies with ANSI N42.18-1980, "Specification and Performance of On-Site Instrumentation for Continuous Monitoring Radioactivity in Effluents."

The staff concluded that this response was acceptable given the fact that the 1980 version of ANSI N42.18 does not differ appreciably from the current version. on this basis, the staff closed RAI-SRP11.5-CHPB-02 and RAI-SRP11.5-CHPB-03.

11.5.2.3 Offline to Inline Monitors for Wastewater Discharge

The applicant changed the radiation monitor from an offline monitor to an inline monitor for the wastewater discharge. The staff concluded that the use of an inline monitor is preferable to an offline type. In addition, the applicant found an inconsistency between the descriptions of the action of the radiation monitor for the wastewater discharge between AP1000 DCD Sections 9.2.9 and 11.5.2. To make Section 11.5.2 consistent with the design description in Section 9.2.9, the applicant eliminated the statement that this monitor controls the basin transfer pumps. The staff concluded that this change is acceptable because Section 11.5 now reflects the actual design and function of the monitor originally described in Section 9.2.9.

11.5.2.4 Changes to Table 11.5-1

The staff reviewed the changes to Table 11.5-1. The staff found the changes acceptable for the following reasons:

- the additional monitors provide improved monitoring capabilities for the main steam line

- the addition of argon (Ar-51) and nitrogen (N-13 and N-16) monitoring in the containment atmosphere improve the monitoring capabilities of this system

- the steam generator blowdown monitors have improved sensitivity

- the fuel handling, auxiliary building, and annex building exhaust vent monitors have improved sensitivity

- the changes to the nominal ranges for main control room air supply duct, containment air filtration exhaust, HP and hot machine shop exhaust, radwaste building exhaust, and the gaseous and liquid radwaste discharge do not change the overall ability of the monitors to perform their intended function

11.5.2.5 Changes to Table 11.5-2

The staff reviewed the changes to Table 11.5-2. The staff concluded that the additional monitoring and location specificity improves the overall area monitoring and is thus acceptable. However, the staff required clarification on the change that removed the rail car bay area as a service area for monitor RMS-JE-RE013. This action seemed to leave the rail car bay area unmonitored. The staff issued RAI-SRP11.5-CHPB-04 to obtain clarification.

The applicant responded that it did not remove the monitor from the rail car bay area but renamed the area. The applicant proposed to revise the title of the area monitor to be consistent with the title of the area. This response is acceptable since the rail car area remains monitored.

In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

11.5.2.6 Additional NUREG-0800 Section 11.5 Acceptance Criteria

Section 11.5 states that the radiation monitoring system is designed in accordance with ANSI N13.1-1969. This standard was withdrawn and replaced in 1999 because the approach taken in the 1969 standard does not provide assurance that the sample in the effluent vent would be representative. The 1999 revision to ANSI N13.1 differs significantly from the earlier version in that it is now performance based. NUREG-0800 Section 11.5 (2007) uses the 1999 standard as acceptance criteria.

The 1969 standard does not provide assurance that the sample from the effluent vent is representative of the particulate matter and reactive vapors passing through the vent. The ability to obtain a representative sample is important since it supports the ability of the licensee to determine public and occupational exposure to radioactivity. The staff concluded that the applicant should use the new standard to ensure that the measurements are accurate. The staff issued RAI-SRP11.5-CHPB-01 to the applicant requesting a change to the newer standard.

On May 22, 2009, the applicant responded to the RAI by stating that committing to the 1999 standard would introduce an excessive degree of uncertainty into the detailed design and construction of the AP1000. To address the limitations of the 1969 standard the applicant proposed an alternative approach relying on best design practices and performance testing and criteria similar to the 1999 standard.

The staff reviewed the response and found that the applicant adopted all relevant performance tests and criteria in the 1999 standard. Once built, the applicant will test the system to ensure that it meets the performance objectives of the 1999 standard. The staff finds this response acceptable.

In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff also reviewed Change Number 45 to AP1000 DCD Tier 2 (proposed DCD Revision 18), Appendix 1A, Sections 3.1.4, 3.6.3.3, 5.2.5.3.3, and 11.5.2.3.1, and LCO 3.4.9

Sections B.3.4.7 and 3.4.9 and as a result issued RAI-SRP11.5-CHPB-05. The applicant subsequently provided sufficient information to demonstrate that the newly proposed Fluorine-18 (F-18) particulate radiation monitor (PSS-JE-RE027) is capable of detecting the AP1000 technical specifications for the reactor coolant system (RCS) leak rate. The discussion of the staff's review and analysis is included in Chapter 23 of this report, Section 23.F,"Changes to Reactor Coolant Pressure Boundary Leakage Detection."

Based on its review and independent verification, the staff concludes that the proposed monitor is sufficiently sensitive to detect the technical specification leak rate. In Revision 18 to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue. On this basis, RAI-SRP11.5-CHPB-05 is resolved.

11.5.3 Conclusion

In NUREG-1793 and its Supplement 1, the staff documented its conclusion that the AP1000 design and DCD (up to and including Revision 15 of the DCD) was acceptable and that the applicant's application for the design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of AP1000 DCD, Section 11.5, "Radiation Monitoring," the staff identified acceptance criteria based on NUREG-0800 Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems." The staff reviewed the AP1000 radiation monitoring and sampling systems for compliance with these requirements, as referenced in NUREG-0800 Section 11.5 and determined that the monitoring systems, as documented in AP1000 DCD, Revision 15, were acceptable because the design conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to AP1000 DCD Section 11.5. The staff concluded that the applicant's proposed changes did not affect the ability of the AP1000 radiation monitoring instrumentation and sampling systems to meet the applicable acceptance criteria in NUREG-0800 Section 11.5. The staff also concluded that the design changes have been properly incorporated into the appropriate sections of the AP1000 DCD. On the basis that the AP1000 monitoring system for process and effluents continues to meet all applicable acceptance criteria and the changes are properly documented in the updated AP1000 DCD, the staff concludes that all of the changes related to the system design of the AP1000 process and effluent radiation monitoring are acceptable.

12. RADIATION PROTECTION

12.1 Introduction

The AP1000 design control document (DCD) Tier 2, Chapter 12, "Radiation Protection," Revisions 16 and 17 include changes to the descriptions of the commitments pertaining to the radiation protection measures and programs of the AP1000 design, as described in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004. As such, this supplemental document to NUREG-1793 must be used in concert with the original issue of NUREG-1793 to completely understand the full evaluation of the AP1000 standard design.

In the AP1000 design certification amendment, the applicant provided additional information related to the Radiation Protection Program and the design features that will ensure that occupational radiation exposures are as low as is reasonably achievable (ALARA). It also provided information on related facility design changes submitted in various technical reports (TRs) potentially affecting the internal and external radiation exposures to station personnel, contractors, and the general population, resulting from plant conditions, including anticipated operational occurrences that will be within regulatory criteria. The staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed the additional and amended information provided by the applicant, using the guidance in Chapter 12 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued March 2007. The NRC developed the original NUREG-1793 using the guidance from regulatory guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, issued November 1978. Therefore, this supplement follows the format of the original NUREG-1793.

The scope of this review includes Chapter 12, Sections 12.1 through 12.5, Revision 17 of the AP1000 DCD, Tier 2 and Tier 1, and Section 3.5, "Radiation Monitoring," which covers the associated inspections, tests, analyses, and acceptance criteria (ITAAC).

Each section of this report describes the staff's evaluation and review results of the changes proposed in Chapter 12 of the AP1000 DCD, Revision 17. The staff reviewed Revision 16 upon receipt and after review issued several requests for additional information (RAIs); however, the applicant submitted Revision 17 prior to the resolution of these RAIs.

12.1.1 Compliance with Title 10 of the *Code of Federal Regulations* Part 20, "Standards for Protection Against Radiation"

The applicant has submitted several TRs with radiation protection implications that include references to industry standards and other regulatory guidance. The applicant's document, TR-98, "AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-098, Revision 0, 'Compliance with 10 CFR 20.1406' " (APP-GW-GLR-017), references Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1406, "Minimization of contamination." The applicant has documented changes to the radwaste building in TR-116, "Additional Liquid Radwaste Monitor Tanks and Radwaste Building Extension" (APP-GW-GLN-116), Revision 0. The applicant documented structural changes in TR-54, "Spent Fuel Storage Racks Structure/Seismic Analysis" (APP-GW-GLR-033), Revision 0, and a redesign of the reactor vessel (RV) head in

TR APP-GW-GLE-016, “Impact of In-Core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package,” Revision 0.

12.1.2 Compliance with 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”

Based upon the discussion in Section 12.1 of NUREG-1793, the standardized power plant designer or combined license (COL) applicant will satisfactorily demonstrate that the radiation protection measures incorporated in the AP1000 program, as documented in the DCD, will offer reasonable assurance that, during all plant operations, the occupational doses will be maintained ALARA and within the limits of 10 CFR Part 20. The following sections present the basis for the staff’s conclusions.

12.2 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable

12.2.1 Summary of Technical Information

This section addresses the design, construction, and operations policies to maximize the incorporation of both design and construction engineering practices and industry lessons learned to achieve the desired ALARA objectives.

The applicant revised two areas discussing the COL applicant’s management commitment and compliance with 10 CFR Part 20 and RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants”; RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable”; and RG 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable,” to reflect the plant’s staffing and organizational differences. These changes are editorial in nature and are incorporated in AP1000 DCD Section 12.1.3 as COL actions requiring the COL applicant to provide such information.

The staff determined that all changes in Revision 17 to DCD Section 12.1 are editorial, with the exception of the following item:

> In DCD Section 12.1.2.4, the applicant added statements certifying compliance with 10 CFR 20.1406. These changes are documented in Westinghouse TR-98, “Compliance with 10 CFR 20.1406” (APP-GW-GLN-098), Revision 0.

During its evaluation and confirmation, the staff identified insufficient information relating to the description of design features concerning the compliance with 10 CFR 20.1406. In the process of the staff developing RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” issued June 2008 (formerly DG-4012), the applicant submitted TR-98. This report was intended to identify and justify standard changes to be incorporated in the DCD. Revision 17 to the AP1000 DCD has incorporated the information in the various sections of the DCD, including Chapters 9, 11, and 12. The implementation of the regulations in 10 CFR 20.1406 affects other structures, systems, and components (SSCs) described in the DCD. Even though this information was not described in the other DCD chapters, it was referenced in the response to RAI-SRP12.1-CHPB-01.

The evaluation below provides an overview of RAI-SRP12.1-CHPB-01, and discusses the applicant’s response and the staff’s evaluation of the response.

12.2.2 Evaluation

This section of the DCD lists specific equipment, as well as facility layout and general design considerations, for 10 CFR 20.1406. The description is substantial and provides examples of design features or considerations for meeting 10 CFR 20.1406. These design features appeared to be based on the draft guidance issued for public comment in the development of DG-4012. The staff has since published the guidance as RG 4.21, Revision 0.

The information presented in AP1000 DCD Tier 2, Sections 12.1 through 12.5, Revision 17, identifies some AP1000 general design features that would minimize the contamination of the facility and environment, as well as the generation of radioactive waste. Specifically, DCD Section 12.1.2.4 describes piping and fuel pool design features to comply with 10 CFR 20.1406. However, this information did not address design features that are unique to system designs or their locations in the plant, warranting more technical details. The applicant did not identify those that should be considered as COL action items. The staff asked the applicant to provide this information in RAI-SRP12.1-CHPB-01. In a letter dated September 9, 2008, the applicant described specific features that are incorporated into the AP1000 design to comply with the requirements of 10 CFR 20.1406 or referenced in TR-98, Revision 0, for the systems that were listed in RAI-SRP12.1-CHPB-01 (and for any other plant systems that may generate or transfer radioactive materials or waste).

The staff determined that the analysis provided by the applicant was insufficient because the applicant's response to RAI-SRP12.1-CHPB-01 and TR-98 failed to address specific information regarding compliance with 10 CFR 20.1406. Section 9.4, "Air-Conditioning, Heating, Cooling, and Ventilation System," covers design features for heating, ventilation, and air conditioning systems (HVAC). Additional information was needed concerning design features provided for these systems to prevent or minimize contamination of the environment. For example, whether there are provisions to monitor and collect condensate that may form at coolers or in HVAC ducts that may contain or potentially contain contamination. The staff issued RAI-SRP12.1-SPCV-01 and RAI-SRP12.1-SPCV-02 to request this information from the applicant. This was identified as an open item in the SER with open items and was tracked as Open Item OI-SRP12.1-CHPB-01.

In a letter dated August 19, 2009, the applicant submitted its response to RAI-SRP12.1-SPCV-01 (R1) and RAI-SRP12.1-SPCV-02 (R1). The applicant provided clarifying information regarding the AP1000 design features used to prevent and mitigate the spread of contamination due to HVAC operation or abnormal conditions. Information was provided in the applicant's response on the features to prevent water entry into HVAC ducting from the liquid radwaste system (WLS) and radioactive waste drain system (WRS). The applicant proposed to revise the following DCD sections: 9.4.3.2.1.1, "Auxiliary/Annex Building Ventilation Subsystem"; and 11.4.2.2.1, "Spent Resin Tanks." The staff concludes that the information provided describes the design features that will prevent and mitigate the spread of contamination through ventilation subsystems and is, therefore, appropriate. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text; therefore, RAI-SRP12.1-SPCV-01 and RAI-SRP12.1-SPCV-02 and the associated Open Item OI-SRP12.1-CHPB-01 are resolved.

The staff also asked the applicant to provide information in RAI-TR98-CHPB-01, RAI-TR98-CHPB-02, and RAI-TR98-CHPB-03. In a letter dated December 19, 2007, the applicant described other specific design features that are incorporated into the AP1000 design to comply with the requirements of 10 CFR 20.1406 and that were referenced in TR-98, Revision 0. The applicant has not committed to RG 4.21, Revision 0, but has described the design features in the licensing basis, the referenced TR-98, and in the response to the aforementioned RAIs.

AP1000 DCD Tier 2, Revision 17, Chapter 12, identified some AP1000 general design features that would minimize the contamination of the facility and environment and would minimize the generation of radioactive waste. However, this information did not address design features that are unique to the auxiliary steam and condensate systems. Current generation plant operating experience has demonstrated that normal pressurized-water reactor (PWR) operation will likely result in detectable levels of tritium as well as other radionuclides in the condensate, auxiliary steam system, and boilers. This was identified as RAI-SRP12.1-CHPB-02. The applicant was requested to describe the design features and operating objectives related to the auxiliary steam and condensate transfer system leaks.

In a letter dated March 4, 2010, the applicant submitted its response to RAI-SRP12.1-CHPB-02. In its response, the applicant described that the outdoor piping from the hotwell to the condensate storage tank will be located above grade, such that leakage from the pipe could be detected in the conduct of routine site activities. When describing the interconnected steam and power systems located in the turbine building, the applicant described the collection of leakage via the waste water system and turbine building drains, which are directed to the turbine building sump. This sump requires routine sampling and analysis prior to the discharge into the environment. This is a requirement for the licensed facility operator to demonstrate compliance with the radiological effluent technical specifications (RETS) and Radiological Environmental Monitoring Program (REMP).

Additionally, the applicant described the plant hot water heating system heat exchangers as providing a barrier to the potential spread of contamination, in the event that the auxiliary steam system was to become contaminated by low levels of primary system leakage to the secondary side of the steam generators. The applicant proposed a revision to AP1000 DCD Section 12.1.2.4.1, “Piping,” which describes the configuration of the outdoor piping and collection of turbine building drains.

The staff reviewed the response provided by the applicant and found it to be acceptable in that the response provided a description of the design features provided to address minimization of contamination of the facility that were specifically not addressed in TR-98. The response provides adequate information, which would allow for timely identification of low activity leakage from these sources to preclude contamination of the site, facility or environment where an AP1000 site is located. The description provided addresses operating experience from the operating nuclear power plants, as discussed in DC/COL ISG-006, “Office of New Reactors Interim Staff Guidance on Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications.” In a subsequent revision to the DCD, the applicant made an appropriate change to Tier 2, Section 12.1.2.4.1 of the DCD, which resolves this issue; therefore, RAI-SRP12.1-CHPB-02 is resolved.

12.2.3 Conclusions

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15) were acceptable and that the application for design certification met the requirements of Subpart B, “Standard Design Certifications,” to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design. Based on the information supplied by the applicant, as described above, the staff concludes that the AP1000 design features met the criteria of NUREG-0800 Section 12.1. These design features are intended to maintain individual doses (person-sievert (roentgen equivalent man) (rem)) and total doses to plant workers and to members of the public ALARA, while maintaining individual doses within the regulatory limits of 10 CFR Part 20.

Adding more detailed information to the DCD about the features for minimizing contamination increases standardization of the AP1000 design. Thus, these changes meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

The staff reviewed the applicant's proposed changes to the AP1000 radiation protection section describing activities that ensure that occupational radiation exposures are ALARA, as documented in DCD Section 12.1. The staff finds the applicant's proposed changes to DCD Section 12.1 to be acceptable.

12.3 Radiation Sources

12.3.1 Summary of Technical Information

The staff approved AP1000 DCD, Section 12.2, "Radiation Sources," Revision 15, in the certified design. This review addresses the compliance with 10 CFR Part 20 and General Design Criterion (GDC) 61, "Fuel Storage and Handling and Radioactivity Control," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities."

In the AP1000 DCD, Revision 17, the applicant proposed the following two technical changes with associated TR changes:

(1) In AP1000 DCD Section 11.2, the applicant proposed to increase the overall liquid waste holdup capacity and improve operational flexibility by adding three liquid waste monitor tanks (and associated pumps, piping, instruments, and valves). The applicant proposed to house the additional tanks, 56,775 liters (15,000 gallons) each, in the radwaste building. The new tanks are identical to the three existing monitor tanks, which are housed in the auxiliary building. The applicant documented these changes in TR-116, Revision 0.

(2) In AP1000 DCD Section 9.1.2.1, the applicant proposed to increase the overall capacity of the spent fuel pool (SFP) from storage locations for 619 fuel assemblies to locations for 884 fuel assemblies. The applicant documented these structural changes in TR-54, Revision 0. The applicant documented the heat loading analysis changes in TR-103, "Fluid System Changes" (APP-GW-GLR-019), Revision 2.

The evaluation below discusses these changes and missing information and provides an overview of the staff's RAI-SRP12.2-CHPB-01 and RAI-SRP12.2-CHPB-02, the applicant's response, and the staff's evaluation of the response.

12.3.2 Evaluation

Liquid Waste Hold-up Tank

The staff reviewed all technical changes to the radiation sources identified in the AP1000 DCD, Revision 17, in accordance with NUREG-0800 Section 12.2, “Radioactive Sources.” In addition, the staff reviewed the entire section to ensure that there was no missing information critical to providing adequate protection to public health and safety.

NUREG-1793 documents the regulatory basis for DCD Section 12.2, Revision 15. The staff has reviewed the proposed changes to DCD Section 12.2 against the applicable acceptance criteria in NUREG-0800 Section 12.2. The following evaluation discusses the results of the staff’s review.

In TR-116, the applicant proposed to add three liquid waste monitor tanks (and associated pumps, piping, instruments, and valves) and to house the additional tanks, 56,775 liters (15,000 gallons) each, in the radwaste building.

In its review of AP1000 DCD Section 12.2, the staff identified areas in which additional information was necessary to complete its evaluation of the applicant’s change (RAI-SRP12.2-CHPB-01). In DCD Tier 2, Section 11.2, the staff noted that a potential exists for the quantity of the radionuclides in the radwaste building portion of the liquid waste management system to exceed the A_1 value. The DCD states that, “The monitor tanks in the non-seismic radwaste building are used to store processed water. The radioactivity content of processed water in each tank will be less than the A_1 and A_2 levels of 10 CFR 71, Appendix A, Table A-1 (“A_1 and A_2 Values for Radionuclides”).”

In 10 CFR 71.4, “Definitions,” defines A_1 as the maximum activity of special form radioactive material permitted in a Type A package. A_2 is defined as the maximum activity of radioactive material, other than special form materials, low specific activity, and surface contaminated object material permitted in a Type A package.

The description of the radioactive sources listed for liquid waste tanks included in AP1000 DCD Section 12.2 did not indicate an increase in volume to approximately 170,325 liters (45,000 gallons) and hence an increase in the overall radioactivity that would thereby be a much larger source of occupational radiation to personnel in the radwaste building.

In Tier 2, Figure 12.3-1 (Sheet 14 of 16), “Radiation Zones, Normal Operation/Shutdown Radwaste Building EL 100’-0” indicates that the room (Room No. 50355) that the tanks will be located in is Plant Radiation Zone 1, which is defined by Figure 12.3-1 (Sheet 1 of 16) in Tier 2, as very low or no radiation sources: “Inside Controlled Area” and Outside “Restricted Area.” This is an area that will result in a dose rate of less than or equal to 2.5 microSieverts per hour (μSv/h) (0.25 millirem per hour (mrem/h)).

The applicant responded to the staff’s concerns in a letter dated September 9, 2008. The applicant agreed that the liquid waste tanks could alter the plant radiation zone assignment and that the issue should be re-analyzed. The affected area will be reclassified to a Plant Radiation Zone III. Plant Radiation Zone III is defined in AP1000 DCD Tier 2, Section 12.3, Figure 12.3-1, as being an area of low-to-moderate radiation sources; limited worker occupancy, with maximum design dose rates less than or equal to 150 μSv/h (15.0 mrem/h). The revised radiation zone reflects the potential increase in volume and hence the increased radioactivity stored in the liquid waste monitor tanks. Given the building design, tank thickness and potential radioactivity stored in the liquid waste monitor tanks, assigning the adjacent area as a Plant Radiation Zone III is appropriate. Based on the evaluation of the DCD information and the applicant's response to the RAI, the staff concludes that the applicant properly identified all design information related to the three additional liquid waste monitor tanks and associated equipment and provided an adequate demonstration that design objectives have been met for the contained source terms described in the DCD as the basis for radiation design shielding calculations and personnel dose assessment. Therefore, RAI-SRP12.2-CHPB-01 is resolved.

Spent Fuel Pool Capacity

In AP1000 DCD Section 9.1.2.1, "Design Basis," the applicant increased the overall capacity of the spent fuel storage from the proposed storage locations for 619 fuel assemblies to storage locations for 884 fuel assemblies. The staff noted that the additional fuel assemblies were not addressed in DCD Section 12.2.1.2.3, "Spent Fuel," nor included in Table 12.2-25, "Fuel Handling Area Airborne Radioactivity Concentrations." The potential addition of 265 fuel assemblies with 0.25 percent fuel defects would increase the airborne radioactivity. Moreover, Table 12.2-25, did not identify the basis of the parameters included in Table 12.2-24 for the number of fuel assemblies used in its calculations. The staff presented these concerns in RAI-SRP12.2-CHPB-02 and the applicant responded in a letter dated September 9, 2008. The air activity was based on a full core offload, with RV design head removal at 100 hours after shutdown; completion of core offload was determined to be 10 days (that is, 240 hours after shutdown); and the SFP purification system was assumed to be operating at 946 liters per minute (250 gallons per minute). Table 12.2-25 was based only on the core from the recent full offload and thus not affected by the increase in the number of fuel assemblies in the SFP. The applicant committed to completing a detailed review and revising the response, if necessary. This was tracked as Open Item OI-SRP12.2-CHPB-02 in the SER with open items.

In a letter dated September 25, 2009, the applicant submitted its response to Open Item OI-SRP12.2-CHPB-02, which provided a discussion that defined the basis and calculations made to assess the maximum airborne activity in the fuel handling area. Following the review of the calculations and additional information provided in the response, the staff concludes that the information in the applicant's response on the airborne activity based on a full core off-load is conservative and, therefore, appropriate. RAI-SRP12.2-CHPB-02 and the associated Open Item OI-SRP-12.2CHPB-02 are resolved.

12.3.3 Conclusions

In NUREG-1793 and its Supplement 1, the staff documented its conclusions that the AP1000 design and DCD (up to and including Revision 15 to the DCD) were acceptable and that the applicant's application for design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of the AP1000 DCD Section 12.2, the staff identified acceptance criteria based on the ability of the design to meet the relevant requirements in 10 CFR Part 20, as it relates to limits on doses to occupationally exposed persons in restricted areas, and in the requirements of GDC 61, as it relates to the information about radiation sources provided by the applicant.

The staff considers RAI-SRP12.2-CHPB-01 to be resolved, and has verified that the appropriate change was incorporated in Revision 17 to AP1000 DCD Tier 2, Figure 12.3-1, Sheet 14 of 16, room number 50355. Additionally, the staff considers Open Item OI-SRP12.2-CHPB-02 to be resolved and concludes that the applicant has committed to follow the guidelines of the RGs and staff positions set forth in NUREG-0800 Section 12.3.

12.4 Radiation Protection Design Features

12.4.1 Summary of Technical Information

The staff approved AP1000 DCD Section 12.3, “Radiation Protection Design Features,” Revision 15, in the certified design. This review addresses system design, performance aspects, and ITAAC only of the radiation protection design features.

The staff’s assessment of the revisions listed in Tier 2, Section 12.3, "Radiation Protection Design Features,” includes changes in Tier 1, Section 3.5, “Radiation Monitoring” (ITAAC) and Tier 2, Section 11.5, “Radiation Monitoring.” These changes do not affect the ITAAC scope and acceptance criteria.

In the AP1000 DCD, Revision 17, the applicant proposed to make the following three technical changes. The staff also identified additional changes related to the density of concrete used for shielding purposes and a description of the computer codes used for shielding calculations. These changes are described below:

(1) In AP1000 DCD Section 12.3.2.2.4, “Fuel Handling Area Shielding Design,” the applicant decreased the minimum water depth above the active fuel portion of the assembly. Spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. Revision 15 of the AP1000 DCD stated that the minimum allowable water depths above active fuel in a fuel assembly during fuel handling were 3.05 meters (m) (10 feet (ft)) in the reactor cavity and in the SFP. This limits the dose to personnel on the spent fuel pool handling machine (SFHM) to less than 25 μSv/h (2.5 mrem/h) for an assembly in a vertical position. Normal water depth above the stored assemblies is about 7 m (23 ft), and for this depth the exposure to plant workers is insignificant. In DCD Revision 16, the minimum allowable water depth above active fuel in a fuel assembly during fuel handling was decreased to 2.89 m (9.5 ft). During the review of Revision 17 to DCD Tier 2, Section 12.3.2.2.4, the staff observed that the applicant decreased the overall minimum allowable water depth above active fuel in the reactor cavity and SFP to 2.67 m (8.75 ft) during fuel movement.

(2) In AP1000 DCD Section 12.3.1.1.2, “Common Facility and Layout Designs for ALARA,” and Section 12.3.5.1, “Administrative Controls for Radiological Protection,” the applicant added statements certifying compliance with 10 CFR 20.1406. TR-98, Revision 0, documented these changes. In the DCD sections, the applicant described general practices, such as to minimize the use of embedded pipes to the extent possible, consistent with maintaining radiation doses ALARA. In addition, to the extent possible, pipes will be routed in accessible areas, such as dedicated pipe routing tunnels or pipe trenches, which provide good conditions for decommissioning, and the number of passageways (doors) between the radiological controlled area and the environment has been minimized. When such doors are incorporated, systems of drains and floor and exterior concrete sloping are used to prevent (potentially radioactive) fluid from exiting the buildings, as well as to prevent surface water from entering the buildings. Because

of the potential for adsorption of contaminated fluids, another feature included minimizing the use of concrete block walls in the radiologically controlled areas of the plant. Where such walls are used, they are fully sealed at the ceiling or top of the block to prevent liquid incursion. The applicant added two COL information items where the COL applicant will, in accordance with 10 CFR 20.1406, establish a groundwater monitoring program beyond the normal radioactive effluent monitoring program and will establish a program to ensure documentation of operational events deemed to be of interest for decommissioning, beyond that required by 10 CFR 50.75, “Reporting and Recordkeeping for Decommissioning Planning.” This or another program will include remediation of any leaks that have the potential to contaminate groundwater.

(3) In AP1000 DCD Tier 2, Section 12.3.1.1.1, the applicant proposed changes to the integrated RV head package and quick-loc connectors. The staff noted that the applicant’s description included an integrated head package that combines the head lifting rig, control and gray rod drive mechanisms, lift columns, control rod drive mechanism cooling system, and power and instrumentation. In its review of DCD Sections 12.3 and 12.4, the staff identified areas that needed additional information to complete its evaluation of the applicant’s change including Figure 12.3-1 (Sheet 8 of 16), “Radiation Zones, Normal Operation/Shutdown Nuclear Island,” EL 135’-3” and Table 12-4-12, “Dose Estimate for Refueling Activities.” The description of the change lacked sufficient detail to determine the radiological impact on occupational exposure (RAI-SRP12.3-CHPB-01).

(4) In AP1000 DCD Section 12.3.2.2.9, the applicant decreased the overall assumed concrete density used for shielding design purposes in the spent fuel transfer canal and tube shielding from 147 pounds per cubic foot (lb/ft^3) to 140 lb/ft^3, without an analysis or description of the potential radiological effects.

(5) In AP1000 DCD Section 12.3.2.3, the applicant described computer codes used to determine design and operational dose rates. The DCD Reference 22 identified the point kernel code Microshield 4 as being a personal computer version with a menu guided user interface of the mainframe calculation code QAD. The staff requested additional information to support verification and validation of this computer code as it is used in the AP1000 DCD dose rate calculations.

12.4.2 Evaluation

The staff reviewed all technical changes to the radiation protection design features identified by change marks in the AP1000 DCD, Revision 17, in accordance with NUREG-0800 Section 12.3 - 12.4, “Radiation Protection Design Features.” Descriptions and evaluations of the radiation protection design features in the AP1000 DCD, Revision 15, that were previously approved are not affected by the new changes and were not re-reviewed by the staff. Information presented in the TRs support all technical changes in this section of the DCD.

The staff reviewed the Tier 1, Section 3.5, “Radiation Monitoring,” ITAAC. This section remained substantially unchanged, but the AP1000 DCD, Revision 17 enhanced the airborne radioactivity and area radiation monitors by adding monitors with multiple detectors and revising the title of selected area monitors. The staff’s evaluation of these Tier 1 changes is in Section 11.5 of this report. No additional technical evaluation was required for Section 12.3.4, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation.”

NUREG-1793 documents the regulatory basis for AP1000 DCD Section 12.3, Revision 15. The staff has reviewed the proposed changes to DCD Section 12.3 against the applicable acceptance criteria of NUREG-0800 Section 12.3-12.4. The following evaluation discusses the results of the staff's review.

12.4.2.1 Fuel Handling Area Shielding Design

In AP1000 DCD Section 12.3.2.2.4, spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. According to Revision 15 of DCD Tier 2, Section 12.3.2.2.4, minimum allowable water depths above active fuel in a fuel assembly during fuel handling are 3.05 m (10 ft) in the reactor cavity and SFP. This limits the dose to personnel on the SFHM to less than 25 μSv/h (2.5 mrem/h) for an assembly in a vertical position. Normal SFP water depth above the stored assemblies is approximately 7 m (23 ft); and, for this depth, the exposure to plant workers is insignificant. TR-121, Revision 0, documented several changes and, as a result, the NRC staff questioned the accuracy of the calculated exposure to workers adjacent to the fuel handling area. The response to RAI-SRP12.2-CHPB-02 clarified the issue raised by the NRC staff to show that it was not possible to maintain the original value of 3.05 m (10 ft) of water over the fuel assembly, given the design of the SFP and its handling equipment. The design drawings showed an actual value of 2.59 m (8.5 ft) of water above the active fuel portion of the fuel assembly. The changes submitted in TR-121 changed the water level in the SFP by 0.3 m (12 inches) to 2.89 m (9.5 ft), and dose rate calculations for this water depth showed that the dose to the SFHM operators would be 25 μSv/h (2.5 mrem/h) or less when moving an irradiated fuel assembly while standing on the SFHM. The applicant documented these changes in TR-121, Revision 0. The applicant also responded to the RAI related to TR-121 in a letter dated October 4, 2007, and described assumptions used in its calculations for the exposure of workers adjacent to the fuel handling areas (RAI-TR121-CHPB-01, RAI-TR121-CHPB-03, and RAI-TR121-CHPB-04). The applicant described the potential radiological effects and dose estimates associated with the change in the minimum water level over active fuel in the refueling area and the SFP.

During the review of Revision 17 to AP1000 DCD Tier 2, Section 12.3.2.2.4, the staff observed that the applicant again changed the overall minimum allowable water depth above active fuel in the reactor cavity and SFP from 2.89 m (9.5 ft) to 2.67 m (8.75 ft) during fuel movement. The applicant did not identify the basis of its parameters in Section 12.3.2.2.4 or the reason for this change. The staff issued RAI-SRP12.3-CHPB-02 to request an explanation for the Revision 17 changes to the minimum refueling and SFP water depth.

The staff reviewed the shielding calculation, APP-GW-N2C-006, Revision 2, "Spent Fuel Shielding Evaluation" (Alt Doc # CN-REA-05-55) referenced in the response to the staff's RAI-SRP12.3-CHPB-02. On the basis of this review, the staff had additional questions related to calculations performed for both 259 cm (102 inches(in)) and 267 cm (105 in) above the active portion of the fuel assembly and how the heights of 259 cm (102 in) and 267 cm (105 in) above the fuel assembly correspond to the minimum required water level in the SFP.

APP-GW-N2C-006 also provided the worst-case source, Case D (4.16E+12 γ–cc/sec) from a single fuel assembly at 259 cm (102 in) and 267 cm (105 in) below the water's surface. It appears from the information supplied in the referenced calculation that the source described in Case A (4.26E+12 γ–cc/sec) is more conservative based solely on the dose contribution from the single elevated fuel assembly in the SFP. The staff requested a clarification to determine the basis for selection of Case D versus Case A, as well as a description of design features/access controls to ensure that the dose to the refueling personnel on the fuel handling bridge deck are maintained ALARA during refueling operations. The staff identified this as Open Item OI-SRP12.3-CHPB-02 in the SER with open items.

In a letter dated July 20, 2009, the applicant submitted its revised response to RAI-SRP12.3-CHPB-02. The applicant indicated that the SFP bridge deck, with 2.54 centimeters (cm) (1 inch) of steel shield, would provide adequate shielding during irradiated fuel movement to maintain whole body exposures to personnel on the SFHM to less than 25 μSv/h (2.5 mrem/h). Calculations referenced by the applicant were reviewed and verified to demonstrate the adequacy of the design to maintain exposures ALARA and were found to be conservative. The staff concludes that the combination of shielding, configuration and SFP water level will maintain exposures ALARA, and is, therefore, acceptable. Therefore, both RAI-SRP12.3-CHPB-02 and the associated Open Item OI-SRP12.3-CHPB-02 are resolved.

12.4.2.2 Compliance with 10 CFR 20.1406

During its evaluation of Revision 17 to the AP1000 DCD, the staff identified insufficient information available on the description of design features concerning its compliance with 10 CFR 20.1406 (RAI-SRP12.1-CHPB-01). The staff was in the process of developing RG 4.21, when the DCD applicant submitted TR-98, Revision 0. The staff intended for its report to identify and justify standard changes to be incorporated into the DCD. Revisions 16 and 17 incorporated the information into the various sections of the DCD, including Chapters 9, 11, and 12. The implementation of 10 CFR 20.1406 affects other SSCs described in the DCD, but that information was not sufficiently described in the other DCD chapters. The design features discussed in DCD Section 12.3 and the COL information items, added as a result of TR-98, clarify some aspects of the applicant's compliance with 10 CFR 20.1406 but do not provide the description of the program consistent with the guidance in RG 4.21.

In TR-98, Revision 0, the applicant proposed to comply with the regulation by the selection of design technology. Table TR-98-1, "AP1000 Features Applicable to 10 CFR 20.1406," in TR-98 lists specific examples (Items 19, 22, 23, 24, 25 and 26 in the table) showing how the AP1000 design complies with the portions of 10 CFR 20.1406 dealing with minimizing the generation of waste. The staff has reviewed the items listed in Table TR-98-1 pertaining to the liquid radwaste system and finds that the applicant addressed the issue of minimization of waste generation in 10 CFR 20.1406. The applicant has not committed to RG 4.21, Revision 0, but has described the design features in the licensing basis for the AP1000 to meet 10 CFR 20.1406 requirements. The staff implemented 10 CFR 20.1406 and issued RG 4.21 after Revision 16 to the DCD but before Revision 17. Therefore, the staff is providing documentation requirements for following this RG in correspondence related to Chapter 12 for COL applicants.

In a letter dated September 9, 2008, the applicant described specific features that are incorporated into the AP1000 design to comply with the requirements of 10 CFR 20.1406 or are referenced in TR-98, Revision 0, for the systems that were listed in RAI-SRP12.1-CHPB-01 (and for any other plant systems that may generate or transfer radioactive materials or waste). The staff determined the analysis is sufficient because the responses to RAI-SRP12.1-CHPB-01, as well as TR-98, provide specific information about compliance with 10 CFR 20.1406, and the staff has determined that the applicant considered the applicable design criteria. (See also Section 12.2.2 of this report).

12.4.2.3 Addition of Integrated RV Head Package and Quick-Loc Connectors

In its review of AP1000 DCD Section 12.3, the staff identified areas that needed additional information to complete an evaluation of the applicant's proposed change (RAI-SRP12.3-CHPB-01). In Tier 2, DCD Section 12.3.1.1.1, the staff noted that the applicant's description includes an integrated RV head package that combines the head lifting rig, control and gray rod drive mechanisms, lift columns, control rod drive mechanism cooling system, and power and instrumentation cabling. The applicant also replaced the conventional top-mounted instrumentation ports/conoseal thermocouple arrangement with a combination thermocouple/incore detector system.

In Revision 17 to AP1000 DCD Tier 2, Figure 12.3-1 (Sheet 8 of 16), "Radiation Zones, Normal Operation/Shutdown Nuclear Island," EL 135′-3" indicates that the RV head stand area may be a Plant Radiation Zone V [less than or equal to 10 mSv/h (1 rem/h)] when the RV head is in the stand, which is defined by Figure 12.3-1 (Sheet 1 of 16). In Revision 15, the same drawing indicated that the area for the RV head stand would be a Plant Radiation Zone II [(less than or equal to 25 µSv/h (2.5 mrem/h)]. There were no supporting calculations to show that the integrated RV head package will result in a dose rate of less than or equal to the original RV head configuration, or to show how this change is ALARA.

Table 12.4-12, "Dose Estimate for Refueling Activities," did not change as a result of the addition of the design change implementing the integrated RV head package. The use of an integrated RV head, which has been installed at several current generation facilities, minimizes the time necessary to perform disassembly and reassembly of the RV during refueling outages.

The description of the change to include the integrated RV head package lacked sufficient information to determine if the containment area radiation zones are affected or if the implementation results in an increase or decrease in the refueling dose estimates. The staff identified this as Open Item OI-SRP12.3-CHPB-01 and was tracked in the SER with open items.

In a letter dated January 21, 2010, the applicant submitted its response to Open Item OI-SRP12.3-CHPB-01.

The applicant indicated that the integrated head package would result in several changes to the radiation zones associated with the upper level of containment and the head stand area. In addition, the applicant revised Table 12.4-12 to reflect a net positive effect concerning the reduction of total dose due to refueling processes. The staff concludes that revised exposure rates and total exposure are consistent with industry data and are appropriate. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text; therefore, both RAI-SRP12.3-CHPB-01 and the associated Open Item OI-SRP12.3-CHPB-01 are resolved.

12.4.2.4 Concrete Density for Shielding Design

During the review of Revision 17 to AP1000 DCD Tier 2, Section 2.3.2.2.9, "Spent Fuel Transfer Canal and Tube Shielding," the staff observed that the applicant decreased the assumed overall concrete density for shielding design purposes from 2354.7 kilograms per cubic meter (kg/m^3)(147 lb/ft^3) to 2242.58 kg/m^3 (140 lb/ft^3). The applicant provided no discussion in DCD Chapter 12 (in Revision 17) describing the effect of an approximate 5-percent decrease in the assumed shielding density of the transfer tube on area radiation levels during fuel movement. With the reduction in the concrete density, the applicant did not identify the basis of the parameters included in Section 12.3.2.2.9 or the reason for the change.

The applicant did not describe the radiological exposure consequences for occupationally exposed personnel nor discuss the effect on radiation zoning. The staff asked the applicant to address these concerns in RAI-SRP12.3-CHPB-03. The staff identified this as Open Item OI-SRP12.3-CHPB-03 and was tracked in the SER with open items.

In a letter dated December 15, 2009, the applicant submitted its response to RAI-SRP12.3-CHPB-03.

The applicant indicated that additional dose calculations were performed and they resulted in redesignation of several radiation zones and two rooms being designated as High Radiation Areas. An additional room now will require access controls consistent with an area greater than 1 rem per hour. Regarding the information in the applicant's response on the changes in radiation zones and controls, the staff concludes that the information provided is consistent with regulatory requirements and guidance and is appropriate. In a subsequent revision to the DCD, the applicant made an appropriate change to the DCD text, which resolves this issue; therefore, both RAI-SRP12.3-CHPB-03 and the associated Open Item OI-SRP12.3-CHPB-03 are resolved.

12.4.2.5 Computer Codes Used for Shielding Calculations

In AP1000 DCD Section 12.3.2.3, the applicant described computer codes used to determine design and operational dose rates for the plant areas located on the AP1000 nuclear island. DCD Reference 22 in Section 12.3 identified the point kernel code, Microshield 4, as being a personal computer version with a menu-driven use interface of the mainframe calculation code QAD. The staff issued RAI-SRP12.3-CHPB-04 requesting additional information to support verification and validation of this computer code as it is used in the AP1000 DCD dose rate calculations.

In its December 15, 2009, response to the staff's RAI, the applicant stated that they had reviewed the AP1000 shielding calculations and updated the DCD to reflect the software used in the most recent shielding analyses of the plant. The applicant revised DCD Reference 22 to cite use of the updated Version 6.20 of the Microshield code. Microshield Version 6.20 is based upon data from the American National Standards Institute (ANSI) standard for gamma radiation attenuation and buildup (ANSI/American Nuclear Society (ANS)-6.4.3-1991, "Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials") and is widely used in the nuclear industry. The applicant stated that they have verified this software code by comparing the software results from this code with the results for reference problems provided in ANSI/ANS-6.6.1-1979, "American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants." While Microshield Version 6.20 is used for non-complex geometries, the applicant modified the DCD to state that they use Monte Carlo or discrete ordinate methods for radiation analysis for complex geometries. The staff finds the applicant's response to RAI-SRP12.3-CHPB-04 to be acceptable because the applicant modified the DCD to clarify and better describe the type of computer code software referenced to perform simple gamma radiation shielding calculations for the AP1000 design and also provided verification and validation of this code's output. Based on the applicant's response, RAI-SRP12.3-CHPB-04 is resolved.

12.4.3 Conclusions

Based on the evaluation of the AP1000 DCD information and response to RAI-SRP12.1-CHPB-01, and the resolution of Open Item OI-SRP12.1-CHPB-01 (see

Section 12.2.2), the staff concludes that the applicant properly identified design information related to its compliance with 10 CFR 20.1406 and provided an adequate demonstration that design objectives for equipment necessary to minimize contamination to the environment have been met, in accordance with 10 CFR Part 20.

Based on the evaluation of the DCD information in the applicant's response to RAI-SRP12.3-CHPB-02 and RAI-SRP12.3-CHPB-03, the staff concludes that the applicant has properly identified all design information related to its compliance with 10 CFR Part 20 and with GDC 61, and has provided an adequate demonstration that design objectives have been met for the spent fuel handling equipment, spent fuel transfer canal, and tube shielding necessary to minimize exposures and maintain personnel doses ALARA, in accordance with 10 CFR Part 20.

Based on the evaluation of the DCD information in the applicant's response to RAI-SRP12.3-CHPB-01, the staff concludes that the applicant provided a complete description of how the placement of the integrated RV head package and the revised and associated equipment in the containment building meet the acceptance criteria of NUREG-0800 Section 12.3 - 12.4.

Based on the evaluation of the DCD information in the applicant's response to RAI-SRP12.3-CHPB-04, the staff concludes that the applicant provided an adequate description of the computer codes used in the shielding calculations for the AP1000 DCD and revised the associated DCD Section 12.3.2.3 and Section 12.3.6 and as a result, meet the acceptance criteria of NUREG-0800 Section 12.3 - 12.4.

The staff's evaluation of the proposed changes (the addition of monitors with multiple detectors and the revision of the title of selected area monitors) in Tier 1, Section 3.5, "Radiation Monitoring," in Revision 17 of the DCD is in Section 11.5 of this report.

In NUREG-1793 and its Supplement 1, the staff documented its conclusion that the AP1000 design and DCD (up to and including Revision 15 to the DCD) were acceptable and that the applicant's application for design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluation of AP1000 DCD Section 12.3, the staff identified acceptance criteria based on the ability of the applicant's design to meet the relevant requirements in 10 CFR Part 20, as it relates to limits on doses to persons in restricted areas; 10 CFR Part 50, as it relates to the inclusion of sufficient design information to demonstrate the objectives for equipment facility design features, shielding, ventilation, area radiation, and airborne radioactivity monitoring instrumentation; and 10 CFR Part 70, "Domestic licensing of special nuclear material," as it relates to the design of radiation protection features to ensure adequate safety under normal and abnormal operating conditions. The staff reviewed the AP1000 design for compliance with these requirements, as referenced in NUREG-0800 Section 12.3 - 12.4, and determined that the design of the radiation protection features, as documented in the AP1000 DCD, Revision 15, was acceptable because it conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 radiation protection design features as documented in the AP1000 DCD, Revision 17. The staff finds that the applicant's proposed changes do not adversely affect the ability of the AP1000 design features to meet the applicable acceptance criteria. The staff also concludes that the applicant has properly incorporated the design changes into the appropriate sections of the AP1000 DCD. On the basis that the AP1000 radiation protection design features continue to meet all applicable acceptance criteria and that the updated AP1000 DCD properly documents the changes, the staff concludes that all of the changes related to the radiation protection features in the AP1000 system design are acceptable.

In addition, the radiation protection pertaining to the IHP and Quickloc connectors contribute to the increased standardization of this aspect of the AP1000 design. Therefore, these changes meet the finality criterion in 10 CFR 52.63(a)(1)(vii).

12.5 Dose Assessment

12.5.1 Summary of Technical Information

The staff approved DCD Section 12.4, “Dose Assessment,” in Revision 15 to the certified design. In Revision 17 to the AP1000 DCD there were no technical changes. This review addresses the anticipated occupational radiation exposure from normal operation and anticipated inspections and maintenance.

12.5.2 Evaluation

The staff reviewed all changes to the “Dose Assessment” section in accordance with NUREG-0800 Section 12.3 - 12.4. The staff also reviewed all changes in the AP1000 DCD, Revision 17. Information presented in the TRs support all changes in the DCD.

NUREG-1793 documents the regulatory basis for AP1000 DCD Section 12.4, Revision 15. The staff reviewed the proposed changes to DCD Section 12.4 against the applicable acceptance criteria in NUREG-0800 Section 12.3 - 12.4. The following evaluation discusses the results of the staff’s review.

12.5.2.1 Summary of Changes

The staff reviewed the applicant’s supporting documentation and determined that there were no supporting calculations that showed the integrated RV head package would result in a dose rate of less than or equal to the original RV head configuration, or that the change was ALARA. Table 12.4-12 did not reflect increased dose (person-sievert (person-rem)) as a result of the addition of the design change implementing the integrated RV head package. The use of an integrated head, which has been installed at several current generation facilities, minimizes the time necessary to perform disassembly and reassembly of the RV during refueling outages. The staff asked the applicant to address these issues in RAI-SRP12.3-CHPB-01. This was addressed as part of the applicant’s response to Open Item OI-SRP12.3-CHPB-01 and was found to be acceptable. (See also Section 12.4.2.3 of this report).

The staff concludes that COL Action Item 12.5-1 was not technical in nature and has not changed from the AP1000 DCD, Revision 15.

12.5.3 Conclusions

As described in Section 12.2 of this report, the response to Open Item OI-SRP12.3-CHPB-01 was found to be acceptable and the staff considers Open Item OI-SRP12.3-CHPB-01 to be resolved. In NUREG-1793 and its Supplement 1, the staff documented its conclusion that the

AP1000 design and DCD (up to and including Revision 15) were acceptable and that the applicant's application for design certification met the requirements in Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design. With the exception of the changes addressed in Open Item OI-SRP12.3-CHPB-01, none of the changes described alter or affect the dose assessment in Revision 15 to the AP1000 DCD. Therefore, the staff requires no technical evaluation of the changes described in DCD Tier 2, Section 12.4.

12.6 Health Physics Facilities Design

This section describes the staff's evaluation and findings of the AP1000 DCD, Revisions 16 and 17. The staff originally reviewed Revision 16 and issued several RAIs, but the applicant issued Revision 17 before the staff could close out the RAIs for the previous revision. This review and evaluation, therefore, encompasses both revisions.

12.6.1 Summary of Technical Information

The staff approved Section 12.6, "Health Physics Facilities Design," of the AP1000 DCD, Revision 15, in the certified design. In Revision 17 to the AP1000 DCD, the applicant proposed one change, described below, with associated TR changes. This review addressed system design and performance aspects only of the health physics facilities design.

In DCD Tier 2, Section 9.1.2, "Spent Fuel Storage," and Section 9.1.4, "Light Load Handling System," the applicant proposed to increase the minimum allowable water depth above the active fuel region in a fuel assembly to 2.89 m (9.5 ft) when an assembly is being transferred in a SFHM. The applicant stated that the proposed increase in minimum water depth was sufficient to ensure that the personnel radiation exposures would be limited to less than or equal to 25 μSv/h (2.5 mrem/h) on the SFHM. The applicant has documented these changes in TR-121, Revision 0.

12.6.2 Evaluation

The staff reviewed the technical changes to the health physics facilities design, in accordance with NUREG-0800 Section 12.3 - 12.4. The staff reviewed all changes identified by change marks in the AP1000 DCD, Revision 17. The information presented in the TRs support all of the technical changes in the DCD.

NUREG-1793 documents the regulatory basis for AO1000 DCD Section 12.5, Revision 15. The staff has reviewed the proposed changes to DCD Section 12.5 against the applicable acceptance criteria of NUREG-0800 Sections 12.3 - 12.4 and 12.5. The following evaluation discusses the results of the staff's review.

12.6.2.1 The Results of Spent Fuel Water Level and Dose

In TR-121, the applicant proposed to change the minimum required depth of water above the active fuel region in a fuel assembly to minimize the exposure from direct radiation to personnel operating equipment on the SFHM. In its review, the staff requested additional information providing the dose rate analysis that was based on an actual increase in the water level from approximately 2.59 m (8.5 ft) to 2.89 m (9.5 ft) above the actual fuel in a fuel assembly when in a SFHM. The applicant responded to the RAI related to TR-121 in a letter dated October 4, 2007, and described the assumptions used in its calculations for the exposure of workers adjacent to the fuel handling areas (RAI-TR121-CHPB-01, RAI-TR121-CHPB-03, and RAI-TR121-CHPB-04). The applicant described the potential radiological effects and dose estimates associated with the reduction of the minimum water level over active fuel in the refueling area and the SFP. Revision 16 to the AP1000 DCD incorporated these changes.

In its review of DCD Section 12.5, the staff identified areas in which the additional information provided by the applicant in a letter dated October 4, 2007, determined that the initial depth of water was in error. The SFHM design actually provided 2.59 m (8.5 ft) depth of water over the active fuel in a fuel assembly when initially proposed. In the applicant's response provided, the elevation of the top of the active fuel was unchanged since the SFP water level was increased by 0.3 m (12 inches). This was necessary to ensure that the exposure rates on the bridge deck (where operating personnel would normally be located) were less than 25 μSv/h (2.5 mrem/h). The information provided, and a review of the requisite guidance in NUREG-0800 Sections 12.3 - 12.4 and 12.5, and in RG 1.13, "Spent Fuel Storage Facility Design Basis," Revision 2, allowed the staff to complete its evaluation of the applicant's change. Further discussion on the water lever change is included in Section 12.4.2.1 of this report.

In DCD Tier 2, Section 9.1.4, the staff noted that a potential exists for the movement of active fuel above the required minimum water depth, if the applicant uses an auxiliary hoist in conjunction with a specialized spent fuel handling tool (SFHT) to reach the approximately 25 percent of the SFP rack spaces that are not accessible using the SFHM. The applicant's response to TR-121, RAI-SRP9.1.4-SBPB-04 describes this activity. The applicant's responses to the staff's RAI are discussed below.

Section 9.1.4.3.7 of the AP1000 DCD states that:

> The three fuel handling devices used to lift spent fuel assemblies are the refueling machine, fuel handling machine, and the spent fuel handling tool. Both the refueling machine and fuel handling machine contain positive stops which prevent the fuel assembly from being raised above a safe shielding height.

DCD Section 9.1.4.3.3 invokes the design of the refueling machine for the SFHM; DCD Section 9.1.4.3.1 states that, because of "mechanical or failure tolerant electrical interlocks or redundant electrical interlocks," the "refueling machine is restricted to raising a fuel assembly or core component to a height at which the water provides a safe radiation shield."

The latter statements imply that, when using the SFHT, there are no positive stops to prevent the fuel assembly from being raised above a safe shielding height. The SFHT with an auxiliary hoist will apparently be used for at least 25 percent of the SFP storage cells, based on the information in TR-121. In TR-121, Revision 0, the applicant stated the following:

> …due to the radius of the FHM manipulator mast and the proximity to the SFP walls, approximately 25 percent of the SFP storage cells cannot be serviced by the mast crane. Also, there are instances where fuel inspection and/or fuel repair require the fuel to be moved from the SFP storage racks to the designated fuel inspection or fuel repair workstation. These non-normal fuel transfer operations are performed using the Spent Fuel Handling Tool (SFHT). The SFHT is a long handled tool which latches onto the fuel assembly top nozzle via manually actuated grippers. Lifting of the SFHT and attached fuel assembly is performed using an auxiliary hoist on the FHM.

The DCD does not describe any interlocks related to the movement of fuel assemblies when using the auxiliary hoist.

In a letter dated June 26, 2008, the applicant submitted a response to RAI-SRP9.1.4-SBPB-04 on fuel handling equipment. The response stated that the refueling machine and the SFHM will contain positive stops to prevent the fuel assembly from being raised above a safe shielding point. The SFHT will only be used in conjunction with the refueling machine and the SFHM. The applicant's response also stated that it would revise AP1000 DCD Tier 1, Section 2.1.1, page 2.1.1-1, to limit the lift height of the refueling machine mast and SFHM mast to maintain the minimum required depth of water shielding. DCD Tier 1, Table 2.1.1-1, includes Item 5 of the ITAAC to describe the acceptance criteria for this design commitment. The staff's evaluation of this Tier 1 change is in Section 9.1.4 of this report.

12.6.2.2 Documentation of Compliance with 10 CFR Part 20

The staff finds that these changes do not affect the design and performance aspects of the health physics facilities, as previously reviewed in NUREG-1793, Section 12.6.

12.6.3 Conclusions

In NUREG-1793 and its Supplement 1, the staff documented its conclusion that the AP1000 design and DCD (up to and including Revision 15 to the DCD) were acceptable and that the applicant's application for design certification met the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

In its previous evaluations of AP1000 DCD Section 12.5, regarding the health physics facilities design, the staff identified acceptance criteria based on the ability of the design to meet the relevant requirements in 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," as it relates to limits on doses to persons in occupied areas, and GDC 61, as it relates to the design of spent fuel storage and handling to ensure adequate safety under normal and postulated accident conditions. The staff reviewed the AP1000 health physics facilities design for compliance with these requirements, as referenced in NUREG-0800 Section 12.5, and determined that the design of the health physics facilities, as documented in the AP1000 DCD, Revision 15, was acceptable because it conformed to all applicable acceptance criteria.

The staff reviewed the applicant's proposed changes to the AP1000 health physics facilities design as documented in the AP1000 DCD, Revision 17. The staff concludes that the applicant's proposed changes do not affect the ability of the AP1000 health physics facilities design to meet the applicable acceptance criteria. The staff also concludes that the applicant has properly incorporated the design changes into the appropriate sections of the AP1000 DCD, Revision 17. On the basis that the AP1000 health physics facilities design continues to meet all applicable acceptance criteria and the updated AP1000 DCD properly documents the changes, the staff concludes that all of the changes related to the system design of the AP1000 health physics facilities are acceptable.

13. CONDUCT OF OPERATIONS

13.3 Emergency Planning

13.3.1 Introduction

In Revision 17 to the AP1000 design control document (DCD), the applicant proposed changes to the annex building Technical Support Center (TSC). The specific DCD changes include: (1) renaming the TSC area in the annex building from the Main TSC Operations Area to the Control Support Area (CSA); (2) removing the identification of the specific TSC location from Tier 1 DCD information; (3) identifying the TSC location in the CSA as Tier 2; and (4) providing additional Tier 1 and Tier 2 DCD conforming changes to reflect the new TSC and CSA designations. The technical justification for the proposed changes is provided in technical report (TR)-107, "AP1000 Technical Support Center," APP-GW-GLR-107, Revision 1, dated June 14, 2007.

13.3.2 Regulatory Basis

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 52.48, "Standards for review of applications," the Nuclear Regulatory Commission (NRC) staff reviewed the AP1000 DCD for compliance with the standards set out in 10 CFR 50.47(b)(8), "Emergency plans," and 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.E, as those standards are technically relevant to the proposed generic DCD changes for the TSC area. Associated TSC guidance is in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Sections 13.3, "Emergency Planning," and 14.3.10, "Emergency Planning – Inspections, Tests, Analyses, and Acceptance Criteria," NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0696, "Functional Criteria for Emergency Response Facilities," and Supplement 1 to NUREG-0737, "Requirements for Emergency Facilities." The specific criterion that applies to the changes evaluated in this section is 10 CFR 52.63(a)(1)(iii), in that the proposed changes reduce unnecessary regulatory burden and maintain protection to public health and safety and the common defense and security.

13.3.3 General Description of Facilities

The TSC provides an area and resources for use by personnel providing plant management and technical support to the plant operating staff during emergency evolutions. In addition, the TSC relieves operators of peripheral duties and communications not directly related to reactor system manipulations and prevents congestion in the control room. Revision 17 of the AP1000 DCD identifies the TSC as the Main TSC Operations Area (Room 40403) in the annex building at Elevation 117'-6", adjacent to the passage from the annex building to the nuclear island control room.

In Section 13.3, "Emergency Planning," of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the staff evaluated the TSC information in Revision 15 of the AP1000 DCD and found that it meets the applicable regulatory requirements in 10 CFR 50.47(b)(8) and 10 CFR Part 50, Appendix E, Section IV.E.

10 CFR Part 52, "License, certifications, and approvals for nuclear power plants," Appendix D, "Design Certification Rule for the AP1000 Design," constitutes the standard design certification (DC) for the AP1000 design and incorporates by reference the associated generic DCD, which includes the Tier 1 and Tier 2 information.

As described in TR-107, the name of the AP1000 annex building TSC (i.e., Main TSC Operations Area) is changed to the CSA, and the DCD is modified to reflect the change. In addition, the identification of the specific TSC location is removed from Tier 1 DCD Section 3.1, "Emergency Response Facilities," and Table 3.1-1, "Inspections, Tests, Analyses, and Acceptance Criteria" (ITAAC). However, TR-107 does not affect the TSC functional requirements and criteria, which are retained and must be met regardless of its actual location. The changes are intended to facilitate a combined license (COL) applicant describing a site-specific location for its TSC, independent of the AP1000 annex building, without the need for an exemption to a Tier 1 AP1000 TSC location. These changes will also ease the NRC's review burden by eliminating the need for such individual COL applicants to apply for exemptions from the standardized DCD wording.

The changes in TR-107 provide flexibility, in that they will allow COL applicants who reference the AP1000 certified design to either use the CSA as the TSC (without departing from the DCD), or designate an alternative TSC location in accordance with the change process in 10 CFR Part 52, Appendix D, Section VIII. The CSA is designed so that it may be used as a TSC, if desired. The generic DCD Revision 17 location for the TSC is maintained in the annex building CSA, and was designated as Tier 2* information in DCD Tier 2, Section 18.8.3.5, "Technical Support Center Mission and Major Tasks," rather than Tier 1 information in DCD Section 3.1. The other DCD Tier 1 requirements associated with the TSC are unaffected by this change, and will be subject to the applicable Tier 1 change control process.

The staff's position is that the DCD Section 18.8.3.5 change should be to Tier 2, rather than Tier 2*. (A change from Tier 1 to Tier 2 would be governed by the same regulatory basis, described above, as a change from Tier 1 to Tier 2*.) This is because the NRC has previously used the Tier 2* designation for DCD information where there is a reasonable expectation of a change over the lifetime of the facility (e.g., a fuel change). The nature of the information is such that the NRC must review and approve the proposed change prior to the change being made. As another example, the Tier 2* designation would be appropriate for information relating to detailed design methodologies and evaluation criteria. Such examples could result in design changes where the safety of the completed design may not be readily apparent. In regard to the AP1000 DCD, once the TSC is built, it is unlikely that it will be moved. Thus, the staff concludes that the Tier 2 designation for the TSC location is more appropriate than Tier 2*. For a Tier 2 TSC DCD location designation, a proposed change to the TSC location (i.e., to a location other than the CSA) by an applicant (or licensee) would require a departure from the certified design.

The staff determined that the applicant must change the TSC location designations in the DCD from Tier 2* to Tier 2. The other DCD Tier 1 requirements associated with the TSC are unaffected by this change, and will be subject to the applicable Tier 1 change control process. The staff identified resolution of this issue as Open Item OI-TR107-NSIR-07. In its January 27, 2010, response to Open Item OI-TR107-NSIR-07, the applicant changed the TSC location designation in DCD Section 18.8.3.5 from Tier 2* to Tier 2, and reflected that change in Revision 19 of the AP1000 DCD. For the reasons discussed above, the staff finds the change acceptable, and Open Item OI-TR107-NSIR-07 is, therefore, resolved.

13.3.4 Conclusion

Based on the above evaluation, the staff concludes that the design changes in TR-107, which are reflected in Revision 19 of the AP1000 DCD, are acceptable because the TSC continues to meet the requirements in 10 CFR 50.47(b)(8) and Section IV.E of Appendix E to 10 CFR Part 50.

13.5 Plant Procedures

13.5.1 Summary of Technical Information

In Revision 19 to the AP1000 DCD, the applicant proposed to partially resolve COL Information Item 13.5-1 in TR-70, “Plant Operations, Surveillance, and Maintenance Procedures” (APP-GW-GLR-040), Revision 1 and in the DCD by addressing; normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, maintenance, inspection, test, and surveillance procedures, as well as the procedures that address the operation of post-72-hour equipment. The COL applicant will address operational and maintenance programmatic issues to resolve COL Information Item 13.5-1.

13.5.2 Evaluation

NUREG-0800 Section 13.5.2.1 states that the applicant should describe its program for developing the operating procedures and that the staff will review the applicant's program for developing and implementing the operating procedures. The staff reviewed TR-70, Revision 1, and its associated references, and determined that it described a process to manage the development, review, and approval of these procedures, but did not clearly provide the requested description. The staff submitted request for additional information (RAI)-SRP13-COLP-01 to ask that the applicant clarify the program description for developing and implementing the operating procedures. In a letter dated July 29, 2008, the applicant stated that TR-70, Revision 1, described the program. Subsequent to the submission of TR-70, Revision 0, the staff met with the applicant to discuss procedure development issues related to the AP1000 design and to allow the staff an opportunity to audit a variety of AP1000 operations procedures. TR-70, Revision 1, addresses issues discussed in this meeting, as well as the concerns regarding NUREG-0711, “Human Factors Engineering Program Review Model,” issued February 2004, that were discussed and later addressed in the staff’s letter, “Summary of the April 11 and 12, 2007, Meeting to Discuss AP1000 Plant Operating Procedures,” dated May 11, 2007. Along with TR-70, Revision 1, the applicant submitted the “AP1000 Writer’s Guidelines for the Normal and Two-Column Format Procedures” (APP-GW-GJP-100 and 200). After reviewing the AP1000 Writer’s Guidelines, which clarified procedure development, the staff finds TR-70, Revision 1 acceptable because the applicant’s program description for developing and implementing the operating procedures meets the guidance in NUREG-0800 Section 13.5.2.1.

Similarly, with respect to the development of emergency operating procedures (EOPs), NUREG-0800 Section 13.5.2.1 states that the applicant should describe its program for developing EOPs, as well as the required content of the EOPs, and that the staff will review the applicant's program for developing and implementing the EOPs. The staff reviewed TR-70 and its associated references and determined that it described a process to manage the development, review, and approval of these procedures, but did not clearly provide the requested description. The staff submitted RAI-SRP13-COLP-02 to ask the applicant to describe the program for developing and implementing the EOPs. In a letter dated

July 29, 2008, the applicant stated that TR-70, Revision 1, described the program. Subsequent to the submission of TR-70, Revision 0, the staff met with the applicant to discuss procedure development issues related to the AP1000 design and to allow the staff an opportunity to audit a variety of the AP1000 operations procedures. TR-70, Revision 1, addresses issues discussed in this meeting, as well as the concerns regarding NUREG-0711 that were discussed and later addressed in the staff's letter dated May 11, 2007. Along with TR-70, Revision 1, the applicant submitted the "AP1000 Writer's Guidelines for the Normal and Two-Column Format Procedures" (APP-GW-GJP-100 and 200). After reviewing the AP1000 Writer's Guidelines, which clarified procedure development, the staff finds TR-70, Revision 1 acceptable because the applicant's program description for developing and implementing the EOPs meets the guidance in NUREG-0800 Section 13.5.2.1.

The staff reviewed DCD Section 13.5.1, which references TR-70, Revision 1, and its associated references, which the applicant submitted as a basis for closing this COL action item. During this review, the staff noted that the DCD addressed safety-related logic circuitry but did not specify which organization had responsibility for it. The staff submitted RAI-SRP13-COLP-03 and RAI-SRP13-COLP-04 to ask the applicant to specify which organization had responsibility for safety-related logic circuitry and freeze seals respectively. In a letter dated July 29, 2008, the applicant stated that it had responsibility for these issues, as described in TR-70, Revision 1. The staff finds this acceptable because the applicant's program for developing and implementing the operating procedures includes a complete description of the safety-related logic circuitry and freeze seals and thus meets the guidance in NUREG-0800 Section 13.5.2.1.

The staff reviewed DCD Section 13.5.1, which states that TR-70 partially addresses the requested COL information, which includes normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, maintenance, inspection, test, and surveillance procedures, as well as the procedures that address the operation of post-72-hour equipment, and that the COL applicant will address operational and maintenance issues. The staff was not clear as to which operational and maintenance issues the COL applicant is responsible for or how they differ from those addressed in TR-70. The staff submitted RAI-SRP13-COLP-05 to ask the applicant to clarify these issues. In a letter dated July 29, 2008, the applicant stated that it was responsible for the development, review and approval of normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, maintenance, inspection, test, and surveillance procedures, as well as the procedures that address the operation of post-72-hour equipment, and that the COL applicant was responsible for maintaining those procedures after their approval and acceptance, as well as for operator training in those procedures. The procedures developed and approved by the applicant are generic procedures. Site-specific procedures will be approved and maintained by the COL applicant. The staff finds this acceptable because the allocation for responsibility for developing and implementing the operating procedures meets the guidance in NUREG-0800 Section 13.5.2.1.

The staff reviewed DCD Section 13.5.1, which states that TR-70 submitted several reports to the staff. To the staff, the word "submit" means "docketed"; thus, this is not an appropriate word to use when referring to documents that have not been docketed. The staff submitted RAI-SRP13-COLP-06 to inform the applicant of this meaning. In a letter dated July 29, 2008, the applicant acknowledged the incorrect language and provided corrected text with amended language. The staff finds this acceptable because the applicant's program description now uses language that is clear and consistent with the staff's understanding.

13.5.3 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design changes are acceptable because they meet the guidance in NUREG-0800 Section 13.5.2.1. The proposed DCD changes are acceptable, pursuant to 10 CFR 52.63(a)(1)(vii), on the basis that they contribute to the increased standardization of the certification information.

13.6 Physical Security

13.6.1 Summary of Technical Information

This section of the AP1000 safety evaluation report (SER) documents the staff's review of the physical security aspects of the AP1000 DC application submitted to the NRC by the applicant.

In Revision 19 of the AP1000 DCD Tier 2, Section 13.6, the applicant describes the plant's physical security program, including those elements of physical protection and mitigative measures identified as being within the scope of the applicant's design. The description includes the required physical security elements of a DC application and references TRs that are part of the DC application, on physical protection and mitigative measures. It describes the design for protecting the plant against acts of radiological sabotage; specifically, the plant layout and protection of vital equipment are in accordance with 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," and applicable regulatory guidance. This safety evaluation (SE) incorporates the staff reviews of DCD Tier 2, Revision 19, Section 13.6; DCD Tier 1, Revision 19, Section 2.6.9, and Table 2.6.9-1; the applicant's RAI responses and the ITAAC for the physical security hardware and referenced safeguards TRs.

The applicant responded to several RAIs. The staff found all responses regarding regulatory requirements to be acceptable, including the applicant's responses to the RAIs associated with physical protection and the open items. All RAIs and open items are resolved and discussed in the SE that is designated as including safeguards information (SGI). The DCD and TRs identify vital equipment and vital areas; describe armed responder positions, physical security attributes (e.g., delay barrier(s) within the AP1000 design scope), their characteristics; and analyze adversarial scenarios for design-basis threats (DBTs). Because this information is security sensitive, the comprehensive physical protection SE includes SGI and is not available for public disclosure. Those persons with the correct access authorization and need-to-know may view the SGI version of the physical security SE, hereafter referred to as the "SGI SER of the AP1000," which is located in the NRC's Secure Local Area Network.

The applicant provided the design description and information related to physical protection in the following parts of the DCD: Section 13.6, Section 2.6.9, and referenced safeguards TRs.

In the AP1000 DCD Tier 1, Revision 19 and applicant RAI responses, Section 2.6.9 and Table 2.6.9-1, the applicant describes the design features and ITAAC for physical security hardware, and the design commitments for physical security hardware ITAAC within the scope of the AP1000 design.

In DCD Tier 2, Revision 19, Section 13.6, the applicant states that the "AP1000 Interim Compensatory Measures Report," the "AP1000 Enhancement Report," and the "AP1000 Safeguards Assessment Report" were separately submitted to establish the design of the AP1000 security systems.

In DCD Tier 2, Revision 19, Section 13.6.1, the applicant states that COL applicants referencing the AP1000 design will address site-specific information related to the "Physical Security Plan, Training and Qualification Plan and the Safeguards Contingency Plan," which are the responsibility of the COL applicant.

13.6.1.1 Summary of Technical Information - ITAAC

The applicant provided design-basis information, including associated tables and figures, in accordance with the selection criteria and methodology for developing DCD Tier 1 information, as described in DCD Tier 2, Section 14.3, to support ITAAC for the AP1000 structures, systems, and components (SSCs).

The applicant organized the DCD Tier 1 information in the systems, structures, and topical areas format shown in the DCD Tier 1 Table of Contents. Site-specific structures that are not within the scope of the certified design are to be addressed by the COL applicant that references the AP1000 certified design. Along with design and program descriptions of site-specific physical protection features, the COL applicant is required to address site security provisions during construction of new reactor(s) that is either inside or co-located to an existing PA. In addition, the COL applicant would address, as applicable, the controls and measures necessary for transitioning between new and existing physical protection systems and continue to maintain in effect the site security program and controls required for implementing the protective strategy for operating power reactors. ITAAC are addressed by both the DC and by the COL applicants incorporating its site-specific ITAAC to meet NRC requirements.

The design bases or supporting security analyses and assumptions related to the design descriptions of security-related features incorporated as AP1000 standard design are provided in SGI TR-94, "AP1000 Safeguards Assessment Report," APP-GW-GLR-066. The staff reviewed the DCD Tier 1 information provided by the applicant in accordance with NUREG-0800 Section 14.3.12, "Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria" Revision 1, January 2010.

The applicant provided design descriptions and information related to physical protection systems or features in the following portions of the DCD and referenced TRs:

AP1000 DCD Tier 1, Chapter 2, "System Based Design Descriptions and ITAAC," Section 2.6.9, "Plant Security System," describes the design features and ITAAC for security hardware for the AP1000 design. Table 2.6.9-1, "Inspections, Tests, Analyses, and Acceptance Criteria" describes the design commitments for security hardware that are within the scope of the AP1000 design.

Tier 2, Chapter 1, Section 1.2, "General Plant Description," and Section 1.2.1, "Design Criteria, Operating Characteristics, and Safety Considerations," provide descriptions of the scope of the AP1000 design.

Tier 2, Chapter 13, "Conduct of Operations," Section 13.6, "Security," of the AP1000 describes physical protection systems or features incorporated as a part of the AP1000 standard design. Elements of a site-specific security program such as organization structure, training, operational program implementations, plant procedures, site-specific target sets, protective strategy, design features for security, and fitness for duty program are to be described by the COL applicant, along with an implementing schedule.

Section 14.2.9, “Preoperational Test Descriptions,” identifies tests to be completed prior to operating conditions. Section 14.2.9.1.14, “Class IE DC Power and Uninterruptible Power Supply Testing” and Section 14.2.9.4.13, “Plant Communications System Testing” addresses security components of plant’s lighting and intra-plant communications.

The applicant submitted TR-94, which describes the security measures credited in defending the AP1000 against a DBT in support of the application.

The applicant submitted TR-96, APP-GW-GLR-067, “AP1000 Interim Compensatory Measures Report,” that include information on compliance with the various sections of the Commission Order on Interim Compensatory Measures that was issued to NRC power reactor licensees on February 25, 2002.

The applicant submitted TR-49, APP-GW-GLR-062, “AP1000 Enhancement Report,” which describes design areas as physical security enhancements that will enhance the ability of a COL applicant to meet the general performance objective of 10 CFR 73.55.

The information found in these referenced reports is considered SGI and Official Use Only Security-Related Information and is protected in accordance with 10 CFR 2.390, “Public inspections, exemptions, requests for withholding,” and 10 CFR 73.21, “Protection of Safeguards Information: Performance Requirements.”

The applicant submitted, in Revision 19 of the AP1000 DCD and RAI responses, proposed changes to physical security hardware ITAAC in DCD Tier 1, Section 2.6.9, “Plant Security System.” As a result of the staff’s review, the applicant submitted additional proposed changes to DCD Tier 1, Section 2.6.9 in its responses to RAI-SRP14.3.12-NSIR-06 and RAI-SRP14.3.12-NSIR-07. The final proposed AP1000 physical security hardware ITAAC was provided in the response to RAI-SRP14.3.12-NSIR-07, Revision 1.

13.6.2 Regulatory Basis – Physical Security

The NRC’s regulations for protecting nuclear power reactors in 10 CFR Part 73, “Physical protection of plants and materials,” include specific security and performance requirements that, when implemented correctly, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect SGI against unauthorized release.

Regulations in 10 CFR 73.1(a)(1), “Purpose and scope,” require the establishment of physical protection systems to protect special nuclear material against the DBT for radiological sabotage, and 10 CFR 73.55 describes the required physical protection for licensed activities. Pursuant to 10 CFR 50.34(c)(2), “Contents of applications; technical information,” and 10 CFR 52.79(a)(35)(i), “Contents of applications; technical information in final safety analysis report,” applicants must prepare and maintain security plans that describe the security-related actions they will take to protect their facilities against acts of radiological sabotage.

Subpart B of 10 CFR 52.47, “Contents of applications; technical information,” requires that information submitted for a DC include performance requirements and design information sufficiently detailed to permit an applicant to prepare procurement specifications and construction and installation specifications. According to 10 CFR Part 52.48, the NRC will review applications filed under 10 CFR Part 52 for compliance with the standards set forth in 10 CFR Part 73.

The AP1000 design descriptions, commitments, and acceptance criteria for the security features, including the plant's layout and protection of vital equipment, as described in the DC application, are based on meeting the relevant requirements of the following Commission regulations:

- 10 CFR Part 50
- 10 CFR Part 52
- 10 CFR Part 73, Appendix B, "General Criteria for Security Personnel"; Appendix C, "Nuclear Power Plant Safeguards Contingency Plans"; Appendix G, "Reportable Safeguards Events"; and Appendix H, "Weapons Qualification Criteria"
- 10 CFR 73.1(a)(1)
- 10 CFR 73.70(f), "Records"
- 10 CFR Part 74, "Material control and accounting of special nuclear material"
- 10 CFR 100.21(f), "Non-seismic site criteria"

In its review, the staff used NUREG-0800 Section 13.6.2 to complete its AP1000 physical security DC review. The following paragraphs in 10 CFR 73.55 include acceptance criteria related to the staff's review in accordance with NUREG-0800 Section 13.6.2:

- Section (e) – Physical barriers: The licensee shall locate vital equipment only within a vital area, which, in turn, shall be located within a protected area (PA), such that access to vital equipment requires passage through at least two physical barriers (as defined in 10 CFR 73.2, "Definitions") that perform their required function in support of the licensee's physical protection program. The physical barriers at the perimeter shall be separated from any other barrier designated as a physical barrier for a vital area within the PA. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the PA permit observation. An intrusion detection system detects penetration or attempted penetration of the PA barrier. Isolation zones and appropriate exterior areas within the PA are illuminated. The main control room has bullet-resistant external walls, doors, ceiling, and floors. Vehicle control measures, which include vehicle barrier systems, protect against the threat of assault by land vehicles.

- Section (g) – Access control: The licensee shall control all points of personnel and vehicle access into a PA; this includes providing equipment capable of detecting firearms, explosives, incendiary devices, or other items that could be used to commit radiological sabotage, or a visual and physical search, or both. Unoccupied vital areas are locked and alarmed with activated detection systems that annunciate in both the central alarm station (CAS) and secondary alarm station (SAS) upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the PA) must be isolated within a bullet-resisting structure.

- Section (i) – Detection and assessment systems: All alarms required pursuant to this part must annunciate and display concurrently in at least two continuously staffed onsite alarm stations, at least one of which must be protected in accordance with the requirements of the CAS. The CAS must be inside the PA, and the interior must not be visible from the perimeter of the PA. The applicant must design and equip the continuously staffed CAS and SAS so that a single act cannot disable both. At least one alarm station must maintain the ability to detect and assess alarms, initiate and coordinate an adequate response to an alarm, summon offsite assistance, and provide command and control. The CAS shall be considered a vital area and be bullet resistant, and associated onsite secondary power supplies for alarm annunciators and nonportable communication equipment must be located within vital areas. Alarm devices and transmission lines must be tamper indicating and be self-checking. Alarm annunciation on CAS/SAS computer monitoring stations shall indicate the type of alarm and its location. All emergency exits from protected and vital areas shall be alarmed and secured by locking devices.

- Section (j) – Communication requirements: Each security officer or armed-response individual shall be capable of maintaining constant communications with an individual in each continuously manned alarm station. Conventional telephone and radio- or microwave-transmitted two-way voice communications shall be established with local law enforcement authorities.

- Section (n) – Maintenance, testing, and calibration: Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communications equipment, access-control equipment, physical barriers, and other security-related devices or equipment.

The staff, in its review, used the following regulatory guidance documents:

- Regulatory guide (RG) 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," issued February 1978

- RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," issued April 1998

- RG 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," issued November 1973

- RG 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," issued September 1986

- RG 5.7, "Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas," Revision 1, issued May 1980

- RG 5.44, "Perimeter Intrusion Alarm Systems," Revision 3, issued October 1997

- Information Notice 86-83, "Underground Pathways into Protected Vital Areas, Material Access Areas, and Controlled Access Areas," dated September 19, 1986

- Regulatory Issue Summary 2005-04, "Guidance on Protection of Unattended Openings that Intersect a Security Boundary or Area," April 14, 2005

- "Nuclear Power Plant Security Assessment Format and Content Guide," Information Systems Laboratories, issued September 2007 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML07030054)

- SAND 2007- 5591, "Nuclear Power Plant Security Assessment Technical Manual," Sandia National Laboratories, issued September 2007 (ADAMS Accession Number ML072620172)

The staff evaluates the following specific acceptance criteria for ITAAC:

- 10 CFR 73.1, as it relates to the prescribed requirements for the establishment and maintenance of a physical protection system and for protection against the DBT of radiological sabotage

- 10 CFR 73.55, as it relates to the requirements for physical protection against radiological sabotage of licensed activities in nuclear power reactors

- 10 CFR 73.70(f), as it relates to the requirements specific to alarm annunciation records

- 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC will be built and operated in accordance with the DC; the provisions of the Atomic Energy Act of 1954, as amended (the Act); and NRC regulations

The COL applicant referencing a certified design is responsible for the site-specific security operational programs to meet the requirements in 10 CFR 50.34(c)(2) or 10 CFR 52.79(a)(35)(i) and 10 CFR 52.79(a)(36)(i), (ii), and (iii). This is satisfied, in part, by describing a physical protection system and administrative programs and procedures for implementing a site-specific protective strategy that demonstrates high assurance that the plant is protected against the DBT. The site-specific physical protection system must be reliable and available and must implement defense-in-depth to provide a high assurance of protection. The following specific and performance requirements describe the security operational programs and the physical protection system: 10 CFR Part 26, "Fitness for duty programs"; 10 CFR 73.55; 10 CFR 73.56, "Personnel access authorization requirements for nuclear power plants"; 10 CFR 73.57, "Requirements for criminal history records checks of individuals granted unescorted access to a nuclear power facility or access to Safeguards Information"; 10 CFR 73.70; 10 CFR 73.58, "Safety/security interface requirements for nuclear power reactors"; and 10 CFR Part 74. Regulations in 10 CFR 52.79(a)(36)(i) or 10 CFR 50.34(d) and Appendix C to 10 CFR Part 73 require COL applicants to submit the security program and planning for a safeguards contingency. The performance and specific requirements in Appendix B to 10 CFR Part 73 requires COL applicants to submit a training and qualification plan and implement the training and qualification requirements for readiness of security personnel and responders.

Within this context, the DC applicant must address those elements or portions of physical protection systems that are considered within the scope of the design. However, the DC applicant may include descriptions of security systems or hardware, with supporting technical bases that are beyond the physical configuration for the scope of the design, provided that it is clearly stated that they are within the scope of the DC.

The staff used NUREG-0800 Section 14.3.12, "Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria," to review the applicant's ITAAC submittal. Section 13.6.3.1 of this SER documents the staff's evaluation of these DCD changes.

13.6.2.1 Regulatory Basis - ITAAC

The NRC regulation for protecting nuclear power reactors is provided in 10 CFR Part 73. The regulation includes specific security and performance requirements that, when adequately implemented, are designed to protect nuclear power reactors against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect safeguards information against unauthorized release.

The performance requirements for the physical protection of nuclear power reactors are provided in 10 CFR 73.1(a)(1), which bounds the adversarial characteristics of the DBT, and 10 CFR 73.55. Pursuant to 10 CFR 50.34(c)(2), 10 CFR 50.34(d), 10 CFR 50.54(p)(1), "Conditions of licenses," and 10 CFR 50.54(p)(2), 10 CFR 73.55(c)(4), and as referenced in 10 CFR Part 52, applicants are required to prepare and maintain security plans that describe the security-related actions that they will take to protect their facilities against acts of radiological sabotage.

10 CFR 52.47(b)(1) requires a DC applicant to contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a plant that incorporates the DC is built and will operate in accordance with the DC.

Regulatory requirements and acceptance criteria related to physical protection systems or hardware are, in part, applicable to the DC (i.e., within scope of the design) or may only be applicable to a COL applicant (outside of a DC design scope) and are identified as follows, as specified in NUREG-0800 Section 14.3.12, "Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria" Revision 1, January 2010.

The COL applicant is required to describe commitments for establishing and maintaining a physical protection system (engineered and administrative controls), organization, programs, and procedures for implementing a site-specific strategy that, if adequately implemented, provides a high assurance of protection of the plant against the DBT. The site-specific physical protection system described must be reliable and available and implement the concept of defense-in-depth protection in order to provide a high assurance of protection. The security operational programs and the physical protection system are required to meet specific and performance requirements of 10 CFR Part 26; 10 CFR Part 74; 10 CFR 73.55; 10 CFR 73.56; 10 CFR 73.57; and 10 CFR 73.70. The COL applicant's security program and planning for safeguards contingency are required to meet 10 CFR 50.34(d) and 10 CFR Part 73, Appendix C. The training and qualification program for readiness of security personnel and responders are required to meet performance and specific requirements of 10 CFR Part 73, Appendix B. Within this context, the DC applicant must address those elements or portion of physical protection systems or features that are considered within the scope of the certified portion of the design. The technical basis for physical protection hardware within the scope of the certified portion of the design provides the basis for ITAAC verification and closures. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves the list of ITAAC numbers. Therefore, all ITAAC numbers are consistent with NUREG-0800 Section 14.3.12.

13.6.3 Evaluation – Physical Security

The staff reviewed AP1000 DCD Tier 1, Revision 19, Section 2.6.9 and Table 2.6.9-1; and DCD Tier 2, Revision 19, Section 13.6, applicant RAI responses, and referenced safeguards TRs.

In its review of the referenced safeguards TRs, the staff identified areas in which it needed additional information to complete the review of the applicant's physical security design. The applicant responded to the staff's RAIs as discussed below.

The staff reviewed the applicant's submittals to determine if its consideration of physical security in the AP1000 design was acceptable.

The staff identified several RAIs relating to target sets for the purpose of reviewing the applicant's physical protection program. The applicant provided design details as background information to assist a licensee with the development of site-specific target sets analyses. The staff evaluated the applicant's responses, and found them to be acceptable for the DC review of the AP1000 physical protection program. The applicant stated in TR-94 that target sets were created to aid in the development of the AP1000 physical security system, and that final target sets will be developed by the COL applicant. Upon the completion of its review, the staff determined that the applicant adequately addressed regulations and the NUREG-0800 acceptance criteria that were identified as within the scope of their design.

Combined License Information Items

The staff reviewed the AP1000 description and commitment for the COL information item that COL applicants referencing the AP1000 certified design must address.

Acceptance Criteria

Pursuant to 10 CFR 52.47(b)(1), a DC applicant is required to submit the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC will be built and operated in accordance with the DC, the provisions of the Act, and the NRC's regulations.

In addition to ITAAC, the staff also reviewed the following information that was submitted by an applicant for the physical security design. The following information was provided by an applicant to meet the acceptance criteria identified in Section 13.6.2, "Regulatory Basis," for the physical security DC.

As required by 10 CFR 73.55(e)(9)(i), a DC applicant shall identify vital areas and a list of vital equipment, by location.

As required by 10 CFR 73.55(e)(9)(v) and (vi), a DC applicant shall identify the control room as a vital area and secondary power supply (for alarm annunciator equipment and nonportable communications) as within a vital area.

As required by 10 CFR 73.55(e)(9)(iii), a DC applicant shall provide the design of the locks and alarms of all unoccupied vital areas.

As required by 10 CFR 73.55(e)(5), a DC applicant shall provide the design describing the bullet resistance of the control room and the CAS.

As required by 10 CFR 73.55(g)(1)(i)(B), a DC applicant should identify locks used to protect the facility and special nuclear material as manipulative resistant.

<u>Evaluation of COL Information</u>

The staff evaluated the COL information item identified in its review of the AP1000 DC application and included in DCD Tier 2, Revision 19, Section 13.6.1. COL information items and applicant responses are those physical security requirements from the above six acceptance criteria that are either met partially or are not addressed by the DC applicant. The staff's evaluation determines whether the DC applicant adequately describes those physical security requirements so that a COL applicant would be able to address them during the COL licensing process. The DC applicant need not identify as COL information items those physical security elements required by regulation. However, for physical security elements partially met in the DC application, the DC applicant should explicitly identify which part of the requirement was met and which part the COL applicant referencing the design will be required to meet.

In the COL information item in DCD Section 13.6.1, the applicant states: Combined License applicants referencing the AP1000 certified design will address site-specific information related to the physical security, contingency, and training and qualification plans.

On the basis of the staff review, the COL information item appropriately addresses interface requirements between the referenced AP1000 physical protection system design and the COL applicant.

13.6.3.1 Evaluation – ITAAC

The applicant submitted the following ITAAC for detection and assessment hardware in Revision 17 of the AP1000 DCD Tier 1, Section 2.6.9, "Plant Security System," which addresses ITAAC consistent with NUREG-0800 Section 14.3.12. The numbering system below corresponds to the applicable elements of NUREG-0800 Section 14.3.12.

2. Physical barriers for the PA perimeter are not part of vital area barriers.

3. Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the PA that allows 20 feet of observation on either side of the barrier. Where permanent buildings do not allow a 20-foot observation distance on the inside of the PA, the building walls are immediately adjacent to, or an integral part of, the PA barrier.

4. An intrusion detection system can detect penetration or attempted penetration of the PA barrier.

5. Isolation zones and exterior areas within the PA are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

6. The external walls, doors, ceiling, and floors in the main control room, the CAS, and the last access control function for access to the PA are bullet resistant.

9. An access control system with numbered picture badges is installed for use by individuals who are authorized access to PAs without escort.

10. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the CAS and SAS upon intrusion into a vital area.

11. Security alarm annunciation occurs in the CAS and in at least one other continuously manned station not necessarily onsite.

14. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

After the review of the ITAAC for detection and assessment hardware, the staff determined that the applicant submitted ITAAC within the DCD that are not within the scope of the DC, and that should be submitted as part of a COL application. In order to complete its review, in RAI-SRP14.3.12-NSIR-06, the staff requested that the applicant revise the physical security hardware ITAAC in Tier 1 of the DCD consistent with NUREG-0800 Section 14.3.12.

In its response to RAI-SRP14.3.12-NSIR-06, the applicant proposed to revise DCD Tier 1, Section 2.6.9 to delete any items that are outside the scope of the certified design. The applicant removed ITAAC Items 3, 4, and 9, which will be submitted by COL applicants. The applicant also removed ITAAC Item 2, as the PA barrier will be addressed by the COL applicant in an ITAAC that will be provided for the site-specific design elements of plant security. In addition, the applicant also revised Item 6.

As a result of changes to regulations for "Power Reactor Security Requirements," effective May 26, 2009, in RAI-SRP14.3.12-NSIR-07, the staff requested that the applicant submit revised AP1000 ITAAC that conform to the 10 CFR Part 73 Power Reactor Security Requirements Final Rule.

In its response to RAI-SRP14.3.12-NSIR-07, dated October 15, 2009, the applicant revised the AP1000 physical security hardware ITAAC to be consistent with the Power Reactor Security Requirements Final Rule of May 26, 2009. ITAAC Item 6 was revised to include the SAS. The applicant also revised ITAAC Item 11 to Item 11(a) and included the addition of video assessment capability and the SAS. In addition, the applicant added ITAAC Items 11(b) and 11(c) as follows:

11(b) The central and secondary alarm stations are located inside a protected area, and the interior of both alarm stations is not visible from the perimeter of the protected area.

11(c) The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to assess alarms and communicate with onsite and offsite response personnel.

Based on discussions at an RAI closed meeting on November 24, 2009 between the applicant and the staff, the applicant submitted revised AP1000 physical security hardware ITAAC in its Revision 1 response to RAI-SRP14.3.12-NSIR-07, dated December 16, 2009. The applicant revised ITAAC Item 6 to add the minimum bullet resistance for the Main Control Room, CAS, and SAS. In addition, the applicant revised ITAAC Item 11(a) to add that alarm annunciation and video assessment is displayed concurrently, and that video recording with real time playback capability can provide assessment of activities before and after each alarm. This revision of ITAAC Item 11(a) also reads on video image recording in ITAAC Item 4(b).

On the basis of its review, the staff finds the revised ITAAC for detection and assessment hardware to be acceptable because it is in conformance with the staff's definition of physical security hardware ITAAC that is within the scope of the DC, and the ITAAC are sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

The applicant submitted the following ITAAC for delay or barrier design features in Revision 17 of the AP1000 DCD Tier 1, Section 2.69, "Plant Security System," which addresses ITAAC consistent with NUREG-0800 Section 14.3.12:

1.
    a) Vital equipment is located only within a vital area.
    b) Access to vital equipment requires passage through at least two physical barriers.

7. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

8. Access control points are established to:
    a) Control vehicle and personnel access into the PA.
    b) Detect firearms, explosives, and incendiary devices at the PA personnel access points.

13. Security alarm devices including transmission lines to annunciators are tamper-indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.

As a result of its review of the ITAAC for delay or barrier design features, the staff determined that the applicant submitted ITAAC within the AP1000 DCD that are not within the scope of the DC and that should be submitted as part of a COL application. In order to complete its review, in RAI-SRP14.3.12-NSIR-06, the staff requested that the applicant revise the physical security hardware ITAAC in Tier 1 of the DCD in accordance with NUREG-0800 Section 14.3.12.

In its response to RAI-SRP14.3.12-NSIR-06, the applicant proposed to revise the AP1000 DCD Tier 1, Section 2.6.9 by deleting ITAAC Items 8(a) and 8(b). The applicant indicated that ITAAC Items 8(a) and 8(b), access control points, would be submitted by the COL applicants.

As noted above, as a result of changes to regulations for "Power Reactor Security Requirements," effective May 26, 2009, in RAI-SRP14.3.12-NSIR-07, the staff requested that the applicant submit revised AP1000 ITAAC that conform to the 10 CFR Part 73 Power Reactor Security Requirements Final Rule.

In its response to RAI-SRP14.3.12-NSIR-07, the applicant revised the AP1000 physical security hardware ITAAC to be consistent with the Power Reactor Security Requirements Final Rule of May 26, 2009. The applicant revised ITAAC Item 13 to Item 13(a) and added ITAAC Item 13(b) to include the requirement for intrusion detection and assessment systems to provide visual display and audible annunciation of the alarm in both the CAS and SAS.

Based on discussions at an RAI closed meeting on November 24, 2009 between the applicant and the staff, the applicant submitted revised AP1000 physical security hardware ITAAC in its Revision 1 response to RAI-SRP14.3.12-NSIR-07. The applicant revised ITAAC Item 1(b) to state “Access to vital equipment requires passage through the vital area barrier.” Because the applicant considered the PA barrier outside of the design scope, it indicated that the requirement in 10 CFR 73.55(e)(9)(i) for two physical barriers would be the responsibility of COL applicants. The applicant added ITAAC Item 13(b) as follows:

> 13(b) Intrusion detection and assessment systems provide visual displays and audible annunciation of alarms in the central and secondary alarm station.

On the basis of its review, the staff finds the revised ITAAC for delay or barrier design features to be acceptable, because it is in conformance with the staff’s definition of physical security hardware ITAAC that is within the scope of the DC, and sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

The applicant submitted the following ITAAC systems, hardware, or features facilitating security response and neutralization in Revision 17 to the AP1000 DCD Tier 1, Section 2.6.9, “Plant Security System,” which addresses ITAAC consistent with NUREG-0800 Section 14.3.12:

12. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

15. Emergency exits through the PA perimeter and the vital area boundaries are alarmed.

16. The CAS and SAS:

    a) have conventional (landline) telephone service and other communication capabilities with local law enforcement authorities, and

    b) are capable of continuous communications with security personnel.

As a result of its review of the ITAAC systems, hardware, and changes to regulations for "Power Reactor Security Requirements,” effective May 26, 2009, the staff determined that the applicant submitted ITAAC within the AP1000 DCD that are not within the scope of the DC and that should be submitted as part of a COL application. In order to complete its review, in RAI-SRP14.3.12-NSIR-07, the staff requested that the applicant submit revised AP1000 ITAAC that address the 10 CFR Part 73 Power Reactor Security Requirements Final Rule.

In its original and Revision 1 responses to RAI-SRP14.3.12-NSIR-07, the applicant revised ITAAC Item 15 to include the requirement for emergency exits through the vital area boundaries to be equipped with a crash bar to allow for emergency egress.

The applicant also removed reference to emergency egress through the PA perimeter in ITAAC Item 15. As the applicant considers the PA barrier outside the design scope, it will be the responsibility of the COL applicant to complete the requirement in 10 CFR 73.55(e)(8)(iii) for emergency exits through the PA perimeter.

The applicant revised ITAAC Item 16(a) to include communication with the main control room, removed the phrase “other communication capabilities,” and added ITAAC Item 16(c) to include the requirement for continued operability of non-portable communications equipment in the CAS and SAS in the event of the loss of normal power, by independent power sources.

On the basis of its review, the staff finds the revised ITAAC for systems, hardware, and features to be acceptable, because it is in conformance with the staff’s definition of physical security hardware ITAAC that is within the scope of the DC, and sufficient to verify that the hardware, as finally installed and constructed, will function as designed.

The staff identified in its review of the proposed changes to the physical security hardware ITAAC in AP1000 DCD Tier 1 and Tier 2 as described below:

1. The external walls, doors, ceiling, and floors in the main control room, the CAS, and the SAS are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.

3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

4. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the CAS and SAS upon intrusion into a vital area.

5.
   a) Security alarm annunciation and video assessment information is displayed concurrently in the CAS and the SAS, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter barrier.

   b) The CAS and SAS are located inside the PA, and the interior of each alarm station is not visible from the perimeter of the PA.

   c) The CAS and SAS are designed and equipped such that, in the event of a single act, in accordance with the DBT of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel.

6. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

7.
   a) Vital equipment is located only within a vital area.
   b) Access to vital equipment requires passage through the vital area barrier.

8. Isolation zones and exterior areas within the PA area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

9. Emergency exits through the vital area boundaries are locked, alarmed and equipped with a crash bar to allow for emergency egress.

13.
   a) The CAS and SAS have conventional (landline) telephone service with the main control room and local law enforcement authorities.

   b) The CAS and SAS are capable of continuous communication with security personnel.

   c) Non-portable communication equipment in the CAS and SAS remains operable from an independent power source in the event of loss of normal power.

15.
   a) Security alarm devices including transmission lines to annunciators are tamper-indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.

   b) Intrusion detection and assessment systems concurrently provide visual displays and audible annunciation of alarms in the CAS and SAS.

16. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

In DCD Tier 1, Revision 19, Table 2.6.9-1, Table 3.3-6, and Tier 2, Section 13.6, "Security," the applicant made appropriate changes that reflect the changes identified in RAI-SRP14.3.12-NSIR-07, Revision 1.

The staff concludes that the applicant has adequately described the Tier 1 physical security hardware ITAAC to be incorporated as part of the standard design. The applicant adequately described the plant layout and protection of vital equipment in accordance with the requirements of 10 CFR 73.55 and provided the technical bases for establishing a physical protection system for the protection against acts of radiological sabotage. The applicant has adequately described requirements specific to design for alarm annunciation records in accordance with 10 CFR 73.70(f). The applicant has provided adequate descriptions of objectives, prerequisites, test methods, data required, and acceptance criteria for security-related ITAAC for the certification the AP1000 design. Therefore, the staff concludes that the AP1000 ITAAC within the scope of NUREG-0800 Section 14.3.12 are necessary and sufficient to assure that with respect to these ITAAC, if the inspections, tests, analyses are performed and the acceptance criteria met, a facility referencing the certified AP1000 design has been constructed and will be operated in compliance with the DC and applicable regulations.

ITAAC Combined License Information Items

The staff reviewed the applicant's descriptions and commitments for COL action items for physical security hardware ITAAC that must be addressed by a COL applicant referencing the certified design.

In response to discussions with the staff at an RAI closed meeting on November 24, 2009, the applicant, in its Revision 1 response to RAI-SRP14.3.12-NSIR-07, proposed to include a COL action item in the next revision of the AP1000 DCD, Tier 2, Section 13.6.1, for the COL applicant to address the physical security hardware ITAAC listed above. The staff finds this to be acceptable and in conformance with the physical security hardware ITAAC requirements to protect against the DBT of radiological sabotage as stated in 10 CFR 73.1(a).

Based on its review of AP1000 DCD Tier 2, Section 13.6.2, the staff determined that the applicant has submitted an appropriate COL action item for physical security hardware ITAAC. The ITAAC that the applicant identified as the responsibility of the COL applicant meet the requirements of 10 CFR 52.47(b)(1) for a COL holder referencing the AP1000 design to build and operate in accordance with the DC, the provisions in the Atomic Energy Act and NRC regulations.

13.6.4 Conclusion

The staff finds that the applicant considered and provided descriptions of physical security systems or features in the standard AP1000 design to provide or facilitate the implementation of a physical protection system to protect against acts of radiological sabotage. The details of this information are provided in the SGI SER for the AP1000, which is stored in the automated database of the NRC's Secure Local Area Network Electronic Safe. The applicant adequately described the plant layout for physical protection and identifying vital equipment and areas, in accordance with the requirements of 10 CFR 73.55. The staff evaluated the technical bases and assumptions related to ITAAC for physical security hardware and found them to be adequate.

The applicant identified the generic issues in the following documents as being outside the scope of the AP1000 design: Generic letter (GL) 89-007, "Power Reactors Safeguards Contingency Planning for Surface Vehicle Bombs," dated April 28, 1989; GL 91-010, "Explosives Searches at Protected Area Portals," dated August 27, 1991; and GL 91-003, "Reporting of Safeguards Events," dated March 6, 1991. The staff has identified the generic issues in the above documents as outside the scope of the design and finds the applicant's approach acceptable. The staff's review of the design found that the applicant addressed Task Action Plan Item A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage," in an acceptable method.

The staff finds that the applicant has provided reasonable assurance that the standard AP1000 design, if implemented correctly, will ensure adequate protection against acts of radiological sabotage. The applicant has provided sufficient physical security design information to support the issuance of an amendment to the AP1000 DC.

The staff reviewed AP1000 DCD, Tier 1, Section 2.6.9 and Tier 2, Section 13.6 and the applicant's responses to RAIs issued on Tier 1 and Tier 2 material, performed in accordance with NUREG-0800 Section 13.6.2, and NUREG-0800 Section 14.3.12 Revision 1 of January 2010. Based on its review, the staff determined that the applicant's selection criteria and methodology for the development of Tier 1 information are acceptable, that the implementation of this selection criteria and methodology is appropriate, and that the resultant ITAAC are adequate for verification that a facility referencing the AP1000 design has been constructed and will be operated in compliance with the DC and applicable regulations.

The staff finds that the applicant has adequately described the objectives, prerequisites, test methods, data required, and acceptance criteria for physical security hardware ITAAC for the certification of the AP1000 design.

14. VERIFICATION PROGRAMS

14.2 Initial Plant Test Program

14.2.9 Preoperational Test Abstracts

14.2.9.1 Introduction

Westinghouse (the applicant) has proposed significant changes to the following items in the AP1000 design control document (DCD) related to preoperational testing: (1) squib valves associated with passive core cooling (DCD Section 14.2.9.1.3); (2) control rod drive system (CRDS) (DCD Section 14.2.9.8); and (3) main alternating current (ac) power testing (DCD Section 14.2.9.2.15).

14.2.9.2 Evaluation

The Nuclear Regulatory Commission (NRC) staff sent the applicant a request for additional information (RAI) regarding the passive core cooling system test description in Section 14.2.9.1.3 of the AP1000 DCD. The NRC staff noted in RAI-SRP14.2-CQVP-10, that under the "General Test Methods and Acceptance Criteria," Item (t), the applicant described the testing of the squib valves as they relate to verification of the passive core cooling system safety injection function. The applicant stated that this test does not have to be performed in the plant. The applicant added this last sentence to the test abstract as part of Revision 16 of the AP1000 DCD. In the RAI, the staff requested that the applicant provide justification for this change. In its July 11, 2008, response to this RAI, the applicant stated that the last sentence of this section was added as an editorial change to clarify that this testing could be done without causing the risk of an actual safety injection into the core. Additionally, the applicant stated that the reliability of these valves could be verified without the valves actually being tested in the operating passive core cooling system.

The staff reviewed the applicant's response to this RAI and determined that this change does not affect the test of the squib valves consistent with regulatory guide (RG) 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 3, and the performance of these valves could be adequately verified through qualification testing. On this basis, the staff concludes that the passive core cooling system test adequately addresses proper operation of the squib valves and verifies adequate safety injection, and is, therefore, acceptable. This resolves RAI-SRP14.2-CQVP-10.

In RAI-SRP14.2-CQVP-9, the staff requested additional information regarding the CRDS test description in Section 14.2.9.1.8 of the AP1000 DCD, Revision 16. The staff noted that Section 14.2.9.1.8 of the AP1000 DCD, Revision 16, CRDS, stated that, as a prerequisite for the control rod drive mechanism cooling test, "the plant is at or near normal operating temperature and pressure, and <u>post-core</u> hot functional testing is in progress." The staff noted that the applicant added the word "post-core" to the test abstract as part of Revision 16 of the AP1000 DCD and asked that the applicant justify this change. In its July 11, 2008, response to this RAI, the applicant stated that the addition of the word "post-core" to modify the hot functional testing of the control rod drive mechanism was an editorial change to clarify the fact that this test can only occur after the core is loaded. The staff reviewed the applicant's response to this RAI and determined that this change clarifies the prerequisites of this test, does not affect the test abstract for the CRDS, and is consistent with the test recommended in

RG 1.68. Therefore, the staff finds this change to be acceptable. This resolves RAI-SRP14.2-CQVP-9.

In test abstract 14.2.9.2.15, "Main AC Power System Testing," the staff noted that the applicant included additional verification activities under "General Methods and Acceptance Criteria" for the bus transfer schemes as part of the test activities associated with the main ac power system. The applicant modified this test abstract to ensure that appropriate testing of the bus transfer schemes occurs during the preoperational phase of the initial test program. Section 8.3.1 of the AP1000 DCD provides details regarding the ac power system and the function of the bus transfer schemes.

The staff reviewed the applicant's proposed change to test abstract 14.2.9.2.15 and determined that this change provides a means to verify proper operation of the automatic and maintenance bus transfer schemes, does not affect the test of the main ac power system, and is consistent with the guidance in RG 1.68. On this basis, the staff concluded that the change is acceptable.

The staff notes that the applicant introduced several other changes to the preoperational test abstracts in Revision 17 of the AP1000 DCD. Upon review, the staff found that these changes were consistent with plant design changes and equipment naming conventions and have no significant impact on preoperational testing. Therefore, these changes are acceptable.

14.2.9.3 Conclusion

The staff reviewed the proposed changes to the preoperational test program described in Section 14.2.9 of the AP1000 DCD. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Section XI, "Test Control," requires that the applicant establish a test program which will ensure that all of the testing required to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily in service, and will be performed in accordance with written test procedures which incorporate the requirements and acceptance limits in applicable design documents. On the basis of this review, the staff concludes that the proposed changes satisfy the requirements of Appendix B to 10 CFR Part 50, Section XI.

14.3 Tier 1 Information

14.3.2 Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)

A number of changes to ITAAC were proposed in revisions to the AP1000 DCD. The staff's evaluations of the proposed changes to ITAAC (which are Tier 1 information) appear in those safety evaluation report (SER) sections where the subject SSCs are discussed. For example, with regard to changes in AP1000 DCD Tier 1, Table 2.1.1-1 concerning the fuel handling machine, these changes are evaluated in Section 9.1 of this report.

14.3.3 Design Acceptance Criteria (DAC)

Changes to design acceptance criteria (DAC) were proposed in Revisions 16 and 17 of the AP1000 DCD. The staff's evaluations of the proposed changes to DAC (which are Tier 1 information) appear in those SER sections where the subject SSCs are discussed. For example, the Tier 1 information found in AP1000 DCD Tier 1, Section 2.5.2, "Protection and Safety Monitoring Systems," Item 11, addresses the hardware and software development process for the design, testing and installation of instrumentation and control (I&C) equipment. Changes to this Tier 1 information are evaluated in Section 7.2 of this report.

14.3.5 Changes to Tier 1 Information

This section addresses the proposed changes to the AP1000 DCD Tier 1 information in the application for amendment to the design certification (DC), as supplemented. Tier 1 information includes the following:

- definitions and general provisions
- design descriptions
- ITAAC
- significant site parameters
- significant interface requirements

The Tier 1 information is derived from the AP1000 DCD Tier 2 information. This SER evaluates the proposed changes to the Tier 1 information in those SER sections in which the associated changes to the Tier 2 information are located, except as described below.

14.3.5.1 Evaluation

Section 1.1 of the AP1000 DCD Tier 1 provides the current definition of “as-built” as follows:

> As-built means the physical properties of a structure, system, or component (SSC) following the completion of its installation or construction activities at its final location at the plant site.

This definition will be used in implementing the ITAAC verification process. This definition intends that the determination of whether an SSC meets the acceptance criteria in the respective ITAAC be performed at its final, in-place location. This approach meets the intent of the ITAAC requirement in 10 CFR 52.97(b), “Issuance of combined licenses.”

In a letter dated June 13, 2008, the applicant submitted AP1000 DCD Impact Document APP-GW-GLE-007, Revision 0, “ITAAC Changes.” One of the proposed changes to the ITAAC would change the definition of “as-built.” The proposed change would add the following sentence to the definition of “as-built”:

> Determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation provided that subsequent fabrication, handling, installation, and testing does not alter the properties.

This proposed change to the definition of “as-built” concerned the staff because the AP1000 ITAAC were developed with the expectation that verifications would be performed in the final, in-place location of the SSC. Also, the shipping, storage, handling, and installation performed after vendor fabrication and prior to the final, in-place location can damage an SSC. The staff raised these concerns during discussions with representatives of the Nuclear Energy Institute (NEI).

On August 1, 2008, NEI submitted NEI 08-01, Revision 0, “Industry Guidelines for ITAAC Closure Process under 10 CFR Part 52,” a draft of which was the basis for the proposed change to the definition of “as-built” in APP-GW-GLE-007, Revision 0. Section 3.1.4 of NEI 08-01 includes the following statement to clarify the definition of “as-built”:

> Many ITAAC require verification of “as-built” SSCs. However, some of these ITAAC will involve measurements and/or testing that can only be conducted at the vendor site due to the configuration of equipment or modules or the nature of the test (e.g., measurements of reactor vessel internals). For these specific items where access to the component for inspection or test is impractical after installation in the plant, the ITAAC closure documentation (e.g., test or inspection record) will be generated at the vendor site and provided to the licensee.

The staff understands that it may be impossible to perform some ITAAC verifications of an SSC in its final, in-place location. Therefore, the staff agrees with the NEI proposal to modify the definition of “as-built” with the clarification provided by the above statement and additional documentation demonstrating that subsequent fabrication, handling, installation, and testing did not alter the properties of the SSC.

In RAI-SRP14.3-NWE2-01, the staff requested that the applicant incorporate the above clarification into the AP1000 definition of “as-built” since it significantly restricts the completion of ITAAC at a vendor’s site. In its response to RAI-SRP14.3-NWE2-01, dated September 9, 2008, the applicant declined to incorporate the subject clarification into the definition of “as-built.” In a December 9, 2008, revised response to RAI-SRP14.3-NWE2-01, the applicant proposed to incorporate the subject clarification in Tier 2. The staff’s position is that this approach would provide too much latitude in application of the subject clarification and that the clarification should be incorporated into the definition itself in Tier 1 of the AP1000 DCD. This was identified as Open Item OI-SRP14.3-NWE2-01.

The industry and the NRC subsequently agreed to augment the NEI 08-01 definition of “as-built” to match the definition agreed to between the NRC and the industry at the December 17, 2009, Category 3 public meeting on ITAAC maintenance. The following definition is considered acceptable:

> **As-built** means the physical properties of a structure, system, or component following completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing do not alter the properties.

In its February 19, 2010, response to RAI-SRP14.3-NWE2-01, the applicant adopted the above definition for use in Section 1.1 of Tier 1 of the AP1000 DCD. In addition, in the February 19, 2010, response the applicant indicated that, as the result of an extensive review of the ITAAC in Revision 17 of the AP1000 DCD, to ensure that those ITAAC utilizing the definition of the term “as-built” will be implemented consistent with the new definition, a number of changes to the ITAAC would be necessary. The staff has reviewed the proposed changes to the ITAAC that use the definition of “as-built” and concludes that these changes do not adversely affect inspectability or any technical aspect of the ITAAC and that these changes are acceptable.

The staff confirmed that Revision 19 to the AP1000 DCD contains the acceptable definition of “as-built” and associated ITAAC changes.

14.3.5.2 Conclusion

The proposed change to the DCD, to include the definition of “as-built” and associated changes to the ITAAC, are acceptable.

14.3.6 Design Acceptance Criteria/ITAAC Closure Process

In an April 1, 2010, Revision 1 response to Open Item OI-SRP3.12-EMB-4, the applicant proposed to include Appendix 14.3A in Tier 2, Chapter 14 of the AP1000 DCD as generic guidance for the resolution of DAC in the DCD. In a proposed Tier 2 Appendix 14.3A, the applicant discussed the options for resolving DAC following certification of the design: (1) through amendment of the DC rule; (2) through the combined operating license (COL) application review process; or (3) through ITAAC after COL issuance. For example, for the piping DAC, the applicant proposed that a COL information item be included in the DCD to resolve the piping design outside of the DC amendment. Closure of piping DAC is further discussed in Section 3.12 of this report.

The standard approach outlined above is voluntary on the part of each licensee referencing the standard AP1000 design. The process envisions an NRC review, inspection, or audit of the DAC completion that applies the “one issue, one review, one position” concept as discussed in RG 1.206, “Combined License Applications for Nuclear Power Plants.” Section C.III.5, to DAC resolution for the reference (first) AP1000 plant and to subsequent AP1000 plants. A COL applicant can apply this standard approach to each of the AP1000 design areas that include DAC (i.e., piping design, digital I&C design, and HFE design). When DAC applies, the process indicates the COL applicant is to provide ITAAC and closure schedule indicating the approach to be followed for the site. The staff finds that this standard approach is consistent with the NRC policy for a design-centered-review approach and the regulations and is, therefore, acceptable.

The staff confirmed that Revision 19 to the AP1000 DCD includes the acceptable standard approach.

14.4 Combined License Applicant Responsibilities

14.4.1 Test Specifications and Procedures

14.4.1.1 Introduction

In a letter dated September 22, 2006, and as supplemented in a letter dated May 11, 2007, the applicant submitted technical report (TR)-71A, APP-GW-GLR-037, “AP1000 Test Specifications and Procedures,” for NRC review and approval. The applicant requested resolution of COL Information Item 14.4-2 based on the information provided in TR-71A. Section 14.4 of the AP1000 DCD lists this COL information item and assigns the COL applicant the responsibility of providing test specifications and test procedures for preoperational and startup tests to the NRC for review and approval.

14.4.1.2 Evaluation

In Section 14.4.3 of the AP1000 DCD, Revision 15, the applicant assigned the responsibility for the development of preoperational and startup test specifications and procedures to the COL applicant. Specifically, COL Information Item 14.4-2 states the following:

> The COL applicant is responsible for providing test specifications and test procedures for the preoperational and startup tests, as identified in Section 14.2.3, for review by the NRC.

As part of Revision 16 to the AP1000 DCD, the applicant submitted TR-71A to address COL Information Item 14.4-2. TR-71A outlines the process to be used by the applicant to develop test specifications and draft procedures and provides a list of test specifications and test procedures to be provided in draft form by the applicant to the prospective COL holder.

TR-71A documents the development process for the preparation of 88 preoperational system test specifications and 59 startup test specifications, to be followed by 289 preoperational test procedures and 59 startup test procedures. However, TR-71A does not include the actual test specifications and test procedures for NRC review and approval.

The staff determined that COL Information Item 14.4-2 calls for the actual submittal of test specifications and test procedures by a COL holder to the NRC onsite inspectors for review and approval before as-built systems and plant features are tested in the field. Furthermore, the NRC inspection staff will need to review the actual test specifications and test procedures for components and systems to be tested to verify their acceptability before COL Information Item 14.4-2 can be categorized as resolved. Accordingly, resolution of COL Information Item 14.4-2 will be subject to the NRC's construction inspection program to allow for the necessary plant as-built inspections and walkdowns. On this basis, the staff concludes that COL Information Item 14.4-2 will remain open pending submittal of the required information by the COL holder.

The staff notes that Section 14.4.2 of Revision 17 of the AP1000 DCD includes the following statement:

> The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-037 (Reference 1), and the applicable changes incorporated into the DCD. Test specifications have been developed as indicated in Reference 1 and are available for NRC onsite review at Westinghouse's offices.

The above noted statement is inconsistent with the staff's conclusion; it should be deleted from the AP1000 DCD, and COL Information Item 14.4-2 should be modified to refer the responsibility for the information item to the COL holder. This was designated as Open Item OI-SRP14.2-CQVP-12.

In its September 30, 2009 response to Open Item OI-SRP14.2-CQVP-12, the applicant stated that it would remove reference to TR-71A in a revised version of the AP1000 DCD. The applicant also proposed to remove any references to TR-71A and to restore the text that would make the COL applicant responsible for providing the necessary information. The applicant included the proposed language to modify COL Information Item 14.4-2 in Enclosure 1 of the September 30, 2009 letter. The response to Open Item OI-SRP14.2-CQVP-12 is acceptable.

The staff confirmed that a subsequent revision to the AP1000 DCD includes the correct COL information item.

14.4.1.3 Conclusion

The staff reviewed the information submitted by the applicant in TR-71A to resolve COL Information Item 14.4-2 in Section 14.4 of the AP1000 DCD, Revision 16. On the basis of this review, the staff concludes that COL Information Item 14.4-2 cannot be resolved until after the issuance of the COL. Therefore, COL Information Item 14.4-2 will reflect that required information is to be provided by the COL holder.

14.4.2 Conduct of Test Program

14.4.2.1 Introduction

In a letter dated September 26, 2006, and as supplemented by letters dated May 24, 2007, and June 19, 2008, the applicant submitted TR-71B, APP-GW-GLR-038, "AP1000 Conduct of Test Program," for NRC review and approval. The applicant requested that COL Information Item 14.4-3 be resolved based on the information provided in TR-71B. Section 14.4.3 of the AP1000 DCD lists this COL information item as follows:

> The Combined License applicant is responsible for a startup administration manual (procedure), which contains the administrative procedures and requirements that govern the activities associated with the plant initial test program, as identified in subsection 14.2.3.

14.4.2.2 Evaluation

In Section 14.2.3.1 of the AP1000 DCD, Revision 15, the applicant provided a set of administrative requirements for the conduct of the initial test program. In addition, Section 14.4 of the AP1000 DCD, Revision 15, summarized the COL applicant responsibilities associated with the development of a startup administrative manual (SAM). COL Information Item 14.4-3 required applicants referencing the AP1000 DCD to provide administrative controls for the conduct of the initial test program in the form of a SAM.

As part of Revision 16 of the AP1000 DCD, the applicant submitted TR-71B to resolve COL Information Item 14.4-3. In reviewing TR-71B, the staff noted that the applicant provided a summary overview of the administrative process and program controls to be utilized in the conduct of the AP1000 startup test program at a licensed AP1000 operational plant site.

The staff also noted that TR-71B outlined basic functional relationships, responsibilities, activities, authority, and principles of conduct for the Joint Test Working Group and other organizational groups. Additionally, TR-71B presented a general and informative description of responsibilities and activities related to the testing of power plant equipment in the period between system turnover and plant acceptance.

On the basis of its review of TR-71B, the staff identified several areas that required additional information as presented in RAI-SRP14.2-CQVP-01 through RAI-SRP14.2-CQVP-08 and RAI-SRP14.2-CQVP-11. To this end, the staff requested that the applicant enhance the proposed program description and amplify the administrative requirements in TR-71B consistent with the guidance in RG 1.68 and Section 14.2, "Initial Plant Test Program—Design Certification and New License Applicants," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Specifically, the staff requested that the applicant provide not only a list of activities that will be controlled during the conduct of the initial test program, but also a description of how these activities will be implemented and controlled. The staff also requested that the applicant justify the use of references to other documents in TR-71B that were not currently under review by the staff. In addition, the staff provided the applicant a set of administrative controls that needed to be described as part of the development of the AP1000 SAM.

In its June 19, 2008, response, the applicant provided Revision 2 of TR-71B. The staff reviewed Revision 2 of TR-71B and noted that the applicant had revised the organizational structure in charge of the initial test program and enhanced the description of the administrative controls for the startup testing phase of the initial test program. The applicant also provided partial responses to some of the RAIs and, in certain areas, did not provide a response at all. The applicant stated in its response letter that Revision 2 of TR-71B incorporated the responses to the RAIs and that the initial test program was developed in conformance with RG 1.68, Revision 2, issued August 1978, and Section 14.2 of NUREG-0800, Revision 2, issued July 1981, which is the certified design regulatory basis.

Because COL applicants are incorporating TR-71B by reference in their applications, the staff determined that the applicant's responses created a conflict between the information provided by the applicant and that required to be submitted by COL applicants. In addition, the staff determined that COL Information Item 14.4-3 calls for the actual submittal of a SAM describing the methods and practices that would govern the initial test program at AP1000 sites. This SAM should provide controls for the conduct of the initial test program consistent with the general criteria in RG 1.68, Revision 3, and Section 14.2 of NUREG-0800, Revision 3. On this basis, the staff determined that the existing content and structure of TR-71B does not meet the guidance applicable to COL applicants.

Since the applicant has not provided the necessary information consistent with COL Information Item 14.4-3, the staff concludes that COL Information Item 14.4-3 will remain open in the AP1000 DCD pending submittal of the required information by COL applicants referencing the AP1000 design.

The NRC staff notes that Section 14.4.3 of Revision 17 of the AP1000 DCD includes the following statement:

> The Combined License information requested in this subsection is partially addressed in APP-GW-GLR-038, (Reference 2), and the applicable changes are incorporated into the DCD.
>
> The program management description for the process to develop the AP1000 Startup Administrative Manual is delineated in APP-GW-GLR-038, (Reference 2).

The above-noted statement is inconsistent with the NRC staff's conclusions; it should be deleted from the DCD, and COL Information Item 14.4-3 should be restored to the original COL information item commitment noted in Section 14.4.3 of Revision 15 of the AP1000 DCD. This was identified as Open Item OI-SRP14.2-CQVP-13.

In its September 30, 2009 response to Open Item OI-SRP14.2-CQVP-13, the applicant stated that it would remove reference to TR-71B in a revised version of the AP1000 DCD. The applicant also proposed to remove any references to TR-71B and to restore the text that would make the COL holder responsible for providing the necessary information. The applicant included the proposed language to modify COL Information Item 14.4-3 in Enclosure 1 of the September 30, 2009 letter. The response to Open Item OI-SRP14.2-CQVP-13 is acceptable. The staff confirmed that a subsequent revision to the AP1000 DCD contains the correct COL information item.

14.4.2.3 Conclusion

On the basis of its review of TR-71B, the staff concludes that it lacks the elements that are necessary for a SAM. Furthermore, it is inconsistent with the guidance provided in RG 1.68, Revision 3, and Section 14.2, Revision 3, of NUREG-0800. Therefore, COL Information Item 14.4-3 will remain a responsibility of the COL applicant.

14.4.3 First-Plant-Only and Three-Plant-Only Tests

14.4.3.1 Introduction

In a letter dated June 5, 2006, the applicant submitted TR-6, "AP1000 As-Built COL Information Items," for NRC review and approval. TR-6 identified COL information items that required as-built information or conditions to be completed.

In its request, the applicant proposed a change to COL Information Item 14.4-6, "First-Plant-Only and Three-Plant-Only Tests," in order to clarify that the test requirements apply to a COL holder rather than a COL applicant.

14.4.3.2 Evaluation

COL Information Item 14.4-6 is associated with tests that must be completed only on the first plant or the first three plants. For COL Information Item 14.4-6, the following revision was underlined in TR-6:

> [The COL applicant or holder for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in subsection 14.2.5 or shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to subsequent plants.]*
>
> The Combined License holder will perform the tests or provide the information defined above prior to fuel load.

The staff concluded that a COL holder can only perform these tests after the plant is essentially complete. The staff found this change to COL Information Item 14.4-6 acceptable for five preoperational tests of the nine tests described in AP1000 DCD Section 14.2.5, "Utilization of Reactor Operating and Testing Experience in the Development of Test Program." However, the staff also found that the following tests described in DCD Section 14.2.5 are performed after initial fuel load and during the low-power and power ascension test phase:

- Section 14.2.10.3.6, "Natural Circulation"
- Section 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger"

- Section 14.2.10.4.6, “Rod Cluster Control Assembly Out-of-Bank Measurements”
- Section 14.2.10.4.22, “Load Follow Demonstration”

On the basis of this finding, the staff requested that the applicant revise COL Information Item 14.4-6 to state that the COL holder will perform these tests after initial fuel load and during the low-power and power ascension test phase of the initial test program.

In its February 1, 2007, letter, the applicant revised TR-6 and COL Information Item 14.4-6 to read as follows:

> The Combined License holder(s) for the first AP1000 plant (or first three plants) available for testing will perform the tests defined during the preoperational and startup testing as identified in Subsections 14.2.9 and 14.2.10. Combined License holders referencing the results of the tests will provide the report as necessary. The schedule for providing this information will be provided prior to preoperational testing.

The change proposed by the applicant clarifies the COL holder’s responsibility, in contrast to the previous assignment of responsibility to either the COL applicant or holder, for performing first-plant-only or three-plant-only tests or providing suitable justification for not performing these tests before the start of preoperational testing.

On this basis, the staff determined that this revision to COL Information Item 14.4-6 is acceptable.

14.4.3.3 Conclusion

The staff reviewed the information submitted by the applicant in TR-6 and concluded that the changes adequately clarified the timing of the testing of first-plant-only and three-plant-only tests. On the basis of this review, the staff concludes that the changes proposed by the applicant are acceptable.

15. TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

In Chapter 15, "Accident Analyses," of the design control document (DCD), Westinghouse, (the applicant), described the safety analyses of various design-basis transients and accidents. The results of these analyses demonstrate conformance of the AP1000 design with the acceptance criteria. The acceptance criteria for the design-basis events are based on meeting the relevant requirements and general design criteria (GDC) specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities." The specific acceptance criteria for each event appear in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," issued March 2007. The acceptance criteria include GDC 10, "Reactor Design," pertaining to the specified acceptable fuel design limits, and GDC 15, "Reactor Coolant System Design," pertaining to the design conditions of the reactor coolant pressure boundary (RCPB). These criteria ensure that these limits are not exceeded during any conditions of normal operation, including anticipated operational occurrences (AOOs). For postulated loss-of-coolant accidents (LOCAs), the applicant analyzed various break sizes and locations to show compliance with the emergency core cooling system (ECCS) performance acceptance criteria specified in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

Various sections and tables in Section 15.0 of the DCD describe the transient and accident events analyzed in Chapter 15 and the computer codes used and assumptions made in the safety analyses of these events. The assumptions relate to the design plant conditions, initial conditions, protection and safety monitoring system (PMS) setpoints and time delays to the reactor trip functions, the actuation of engineered safety features (ESFs), and the plant systems and components available for the mitigation of transients and accidents.

The proposed changes to the DCD include numerous changes to Section 15.0. The sections below describe the specific changes and their evaluation by the staff of the U.S. Nuclear Regulatory Commission (NRC).

15.1.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.1.0.3.1 Power Measurement Uncertainty

The initial conditions assumed in the existing Chapter 15 safety analyses include a ±2-percent allowance for the power calorimetric error. In DCD, Section 15.0.3.2, "Initial Conditions," Table 15.0-2, "Summary of Initial Conditions and Computer Codes Used (Sheets 1–5)" (Footnote a), and Table 15.0-5, "Determination of Maximum Power Range Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint and Inherent Typical Instrumentation Uncertainties," now include the following sentence:

> The main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2 percent power uncertainty is conservative.

The applicant also changed the initial thermal power assumed for the large-break LOCA analysis in Table 15.0-2 from 3,468.0 megawatts thermal (MWt) to 3,434.0 MWt. Based on the

AP1000 rated thermal power of 3,400 MWt, the initial thermal power of 3,434 MWt represents a 1-percent allowance for the power calorimetric error.

15.1.0.3.1.1 Evaluation

The review guidance in NUREG-0800 Section 15.0 states that the reviewer ensures that the application specifies the permitted fluctuations and uncertainties with reactor system parameters and assumes the appropriate conditions, within the operating band, as the initial condition for transient analysis. For analyses of postulated LOCAs, Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 specifies that an assumed power level lower than 1.02 times the licensed power level may be used, provided the proposed alternative value has been demonstrated to account for uncertainties resulting from errors in power-level instrumentation.

The existing Chapter 15 safety analyses for all transients and accidents assume the rated thermal power with a calorimetric error of 2.0 percent as the initial condition. The DCD continues to assume the 2 percent power uncertainty for all events except for the large-break LOCA; for this event, it assumes the rated thermal power with a 1 percent power uncertainty as the initial condition. The AP1000 DCD does not describe the instrumentation or methodology used for the main feedwater flow measurement, nor does it provide a basis to support the claimed 1 percent power uncertainty.

In its response to request for additional information (RAI)-SRP-15.0-SRSB-02, the applicant stated that the AP1000 will use the proven application of high-accuracy ultrasonic feedwater flow measurement and high-accuracy feedwater temperature measurement to affect a high-accuracy plant calorimetric. AP1000 licensees will calculate the plant calorimetric uncertainty and verify that the actual plant instrumentation performance is bounded by the design value of 1 percent calorimetric uncertainty. For traceability, Section 15.0.15, "Combined License Information," will include the following combined license (COL) information item:

> 15.0.15.1 Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters prior to fuel load, the Combined License holder will calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

The applicant will also revise DCD Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," to include COL Item 15.0-1, "Documentation of Plant Calorimetric Uncertainty Methodology," as an action item for the COL holder.

The staff notes that the NRC has approved high-accuracy ultrasonic feedwater measurement instrumentation that can achieve a 1 percent power measurement uncertainty, and this instrumentation has been used as a basis for power uprates for operating plants. The staff concludes that COL Information Item 15.0.15.1 and COL Item 15.0-1 provide acceptable vehicles for the COL license holder to confirm its selected instrumentation for the main feedwater measurement, with the power calorimetric measurement uncertainty bounded by the 1 percent power uncertainty assumed in the large-break LOCA analysis.

In a subsequent revision to the AP1000 DCD, the applicant incorporated COL Information Item 15.0.15.1 in DCD Section 15.0.15 and COL Item 15.0-1 in Table 1.8-2.

15.1.0.3.1.2 Conclusion

Based on the above evaluation, the staff finds that COL Item 15.0-1 and COL Information Item 15.0.15.1 provide acceptable vehicles to confirm the 1-percent power uncertainty assumed in the large-break LOCA analysis. The uncertainty pertaining to the initial condition of the power level is, therefore, properly addressed, consistent with NUREG-0800 Section 15.0 and Appendix K to 10 CFR 50.46 regarding LOCAs. Therefore, the staff concludes that the above proposed changes are acceptable

15.1.0.3.2 Axial Power Shape

Section 15.0.3.3, "Power Distribution," in Revision 15 of the DCD states that the axial power shape used in the departure from nucleate boiling (DNB) calculation is the 1.55 chopped cosine, as discussed in Section 4.4 for transients analyzed at full power. In Revision 17 of the DCD, the applicant changed "the 1.55 chopped cosine" to "a chopped cosine."

15.1.0.3.2.1 Evaluation

Section 4.4.4.3.2, "Axial Heat Flux Distributions," of the DCD states that the reference axial shape used in establishing core DNB limits (that is, overtemperature ΔT protection system setpoints) is a chopped cosine with a peak-to-average value of 1.61. The staff finds that the applicant made the proposed change in Section 15.0.3.3 of the DCD to eliminate the specific mention of the 1.55 chopped cosine in order to correct the inconsistency with the value given in Section 4.4.4.3.2 of the DCD. The staff concludes that the proposed change is acceptable because the Chapter 15 safety analyses for the transients were performed with the axial power shape described in DCD Section 4.4.4.3.2.

15.1.0.3.2.2 Conclusion

The staff has reviewed the proposed change in Section 15.0.3.3 of the DCD to eliminate the specific value of 1.55 for the chopped cosine axial power shape. Based on the above evaluation, the staff concludes that this proposed change is acceptable because the applicant made it in order to correct an inconsistency with the value specified in DCD Section 4.4.4.3.2, which was referenced in DCD Section 15.0.3.3.

15.1.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

15.1.0.6.1 Change of "High-1" to "High-2" Containment Pressure for "S" Signal or Engineered Safety Feature Actuation

Revision 17 of the DCD changes Table 15.0-4a, "Protection and Safety Monitoring System Setpoints and Time Delay," from having the "S" signal trip function on high-1 containment pressure to its functioning on high-2 containment pressure. In addition, in Table 15.0-6, "Plant Systems and Equipment Available for Transient and Accident Conditions (Sheet 1 of 5)," the applicant changed the ESF actuation function credited for the steam system piping failure, feedwater system pipe break, and LOCA events from high-1 containment pressure to high-2 containment pressure.

15.1.0.6.1.1 Evaluation

The applicant made the change from “high-1” to “high-2” containment pressure in DCD Tables 15.0-4a and 15.0-6 for the “S” signal trip function and the ESF actuation function credited in the safety analyses, respectively, in order to be consistent with Technical Specification (TS) Table 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” which indicates that the safeguard actuation function 1.b is “containment pressure—high 2.” Although the applicant changed the term “high-1 containment pressure” to “high-2 containment pressure,” the value of the setpoint assumed in the safety analyses is not changed, and therefore the safety analyses are not affected. Since this is merely a change in terminology for consistency, the staff concludes that this change is acceptable.

15.1.0.6.1.2 Conclusion

Under 10 CFR 50.36, “Technical specifications,” the plant TS must specify the limiting safety system settings of automatic protective devices. TS Table 3.3.2-1 for the AP1000 specifies the “Containment Pressure—High 2” trip setpoint for the safeguard actuation function. The staff finds the proposed change from “high-1” to “high-2” containment pressure in DCD Tables 15.0-4a and 15.0-6 to be acceptable because the applicant has made it in order to be consistent with the terminology used in the AP1000 TS, without changing the setpoint value or safety analyses.

15.1.0.6.2 Changes to Pressurizer Water Level Setpoints

In DCD Table 15.0-4a, the applicant changed the limiting setpoint for actuation of passive residual heat removal (PRHR) on the high-3 pressurizer water level from 80 percent of span to 76 percent of span. It also changed the chemical and volume control system (CVS) isolation on the high-2 pressurizer water level from 67 percent of span to 63 percent of span, and the CVS isolation on the high-1 pressurizer water level coincident with “S” signal from 30 percent of span to 28 percent of span.

15.1.0.6.2.1 Evaluation

The applicant changed the high-3, high-2, and high-1 pressurizer water level setpoints because of a design change in pressurizer dimensions. As described in technical report (TR)-36, APP-GW-GLR-016, Revision 0, “AP1000 Standard Combined License Technical Report, AP1000 Pressurizer Design,” the applicant has changed the AP1000 pressurizer dimensions to accommodate the space constraint. This dimensional change reduces the height and increases the diameter of the pressurizer but maintains the same overall pressurizer volume. Therefore, in DCD Table 15.0-4a, the setpoints assumed in the safety analyses for the high-3, high-2, and high-1 pressurizer water levels change from 80 percent, 67 percent, and 30 percent to 76 percent, 63 percent, and 28 percent, respectively, to maintain the same water volumes of the corresponding setpoints. The applicant also revised the high pressurizer water level setpoints in TS Tables 3.3.1-1 and 3.3.2-1, respectively, to be consistent with the revised assumed measurement instrumentation uncertainties accounted for in the safety analysis.

In its response to RAI-TR36-012, the applicant reanalyzed Chapter 15 limiting events using the changed dimension and the revised setpoints. The results show no or minimal effects on the Chapter 15 events, and the applicable acceptance criteria for each event continue to be met. Therefore, the staff concludes that these changes are acceptable.

15.1.0.6.2.2 Conclusions

Based on the above evaluation, the staff finds that the applicant has made the proposed changes to the high-3, high-2, and high-1 pressurizer water level setpoints in DCD Table 15.0-4a and in TS Tables 3.3.1-1 and 3.3.2-1 to accommodate the pressurizer dimension change without changing the corresponding water volume. The staff concludes that these changes are acceptable because the applicant's reanalyses of the Chapter 15 limiting events using the revised dimension and setpoints show no more than minimal effects, and the applicable acceptance criteria for each event analyzed continue to be met.

15.1.0.6.3 Change to the Limiting Setpoint of the Boron Dilution Block on the Source Range Flux Doubling Function

In DCD Table 15.0-4a, the applicant changed the limiting setpoint of the boron dilution block on the source range flux multiplication (doubling) function from "1.6 over 50 minutes" to "3.0 over 50 minutes."

15.1.0.6.3.1 Evaluation

The safety analyses of the boron dilution events during shutdown operation credit the boron dilution block on the source range flux multiplication function (DCD Section 15.4.6, "Chemical Volume and Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant"). In DCD Table 15.0-4a, the applicant changed the terminology for the boron dilution block from "source range flux multiplication" to "source range flux doubling." In Section II.9, "Flux Doubling/Boron Dilution Modifications," of TR-80, APP-GW-GLR-080, "Mark-up of AP1000 Design Control Document Chapter 7," Revision 0, the applicant discussed the change for the setpoint of the source range flux doubling signal for the boron dilution block. The source range flux doubling signal is used for protection against boron dilution events during shutdown operation. If the source range neutron flux were to increase and exceed the setpoint because of a decrease in boron concentration in the reactor coolant, the boron dilution protection functions would be actuated. The applicant stated that an analysis of the source range flux doubling setpoints indicates that there is a significant likelihood that the setpoints could lead to inadvertent actuation of the boron dilution protection actions when the plant is shut down and the flux doubling function is active, even in the absence of actual changes in core neutron multiplication. This occurs because of the variability inherent in counting a discrete random process such as the leakage of neutrons from the reactor core, especially at relatively low count rates. Therefore, Revision 16 of the DCD changed the nominal setpoint from 1.6 over 50 minutes to 2.2 over 50 minutes to reduce the likelihood of inadvertent actuation of the boron dilution protection functions during normal operation.

DCD Section 15.4.6 provides the safety analyses of boron dilution events resulting from CVS malfunction during various modes of operation. The analyses for dilution during cold shutdown (Mode 5), safe shutdown (Mode 4), hot standby (Mode 3), and startup (Mode 2) take credit for the source range flux doubling signal with the increased setpoint that isolates the makeup flow to the reactor coolant system (RCS) from the demineralized water storage tank. The results indicate that automatic protective actions initiate to minimize the approach to criticality and maintain the plant in a subcritical condition.

In Revision 17 of the DCD, TS Table 3.3.2-1, the trip setpoint for function 15.a also specifies the source range flux doubling setpoint of 2.2 over 50 minutes for ESF actuation system function 15, boron dilution block. The staff questioned the value used in the safety analysis with the consideration of instrumentation uncertainties. In its response to RAI-SRP15.0-SRSB-01, the applicant stated that the safety analysis cases reported in DCD Section 15.4.6 support a setpoint of 3.0 over 50 minutes to mitigate the boron dilution event in Modes 3, 4, and 5. It further stated that DCD Table 15.0-4a should also report this value of 3.0 as the setpoint assumed in the safety analysis. The applicant has incorporated this change in DCD, Table 15.0-4a. In its response to RAI-SRP15.4.6-SRSB-02, the applicant stated that the boron dilution analyses performed for Modes 3, 4, and 5 documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes with acceptable results (i.e., the automatic protective actions terminate the boron dilution event and maintain the plant in a subcritical condition). Therefore, the staff concludes that the change in Table 15.0-4a of the DCD, of the limiting setpoint of the boron dilution block on source range flux doubling to 3.0 over 50 minutes, is consistent with the assumption in the safety analysis and, therefore, is acceptable.

Regarding the TS nominal setpoint of 2.2 over 50 minutes, the applicant stated that it has estimated the major factors that contribute to the measurement uncertainty of the boron dilution algorithm. To ensure that the neutron count integrals are as large as possible, the design features include placing the source range neutron detectors outside the reactor vessel along the cardinal axes of the reactor core with the highest neutron leakage, and increasing the counting interval from 60 seconds to 120 seconds in the algorithm. The applicant presumes that the 36 percent difference between the value assumed in the accident analysis and the setpoint selected will bound the value determined when final plant design inputs are available. The staff notes that in DCD Tier 1, Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-8, "Inspections, Tests, Analyses and Acceptance Criteria," design commitment item 10 specifies that the PMS setpoints are determined using a methodology that accounts for loop inaccuracies, response testing, and maintenance or replacement instrumentation. If the measurement uncertainty after the final instrumentation installation is not bounded by the 36-percent allowance, the nominal setpoint would be revised accordingly. Therefore, the staff concludes that there is adequate assurance that the nominal setpoint specified in the TS is consistent with the safety analysis setpoint and the measurement uncertainty.

15.1.0.6.3.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed change of the source range flux doubling setpoint from "1.6 over 50 minutes" to "3.0 over 50 minutes," in DCD Table 15.0-4a is acceptable because this change is consistent with the assumption made in the safety analysis of the boron dilution events with acceptable results.

15.1.0.6.4 Deletion of P-8 Interlock and Replacement with P-10

In Revision 17 of the DCD, the applicant deleted the "High neutron flux, P-8" reactor trip interlock from Table 15.0-4a, and in Table 15.0-6 it changed the permissive interlock for the power range high flux, low-flow trip function credited for the "startup of an inactive reactor coolant pump at incorrect temperature" event from P-8 to P-10.

15.1.0.6.4.1 Evaluation

The power range nuclear power P-8 interlock permits a reactor trip on low flow or reactor coolant pump (RCP) high bearing water temperature in a single loop. In TR-80, the applicant provided the rationale for the deletion of the high neutron flux P-8 interlock from the PMS. As stated in TR-80, Section II.7, "Low Reactor Coolant Flow Reactor Trip Logic Modifications," the

AP1000 is not licensed for N-1 operation and, therefore, the design of the PMS is not required to account for this requirement. The applicant also deleted the P-8 interlock from DCD Table 7.2-3, "Reactor Trip Permissives and Interlocks," and TS Table 3.3.1-1, trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. Therefore, the deletion of P-8 from DCD Table 15.0.4a is consistent with the reactor trip logic and TS. The staff concludes that the deletion of the P-8 interlock from DCD Tables 15.04a and 7.2-3 and TS Table 3.3.1-1 is acceptable because the P-8 interlock is not needed for the AP1000 design.

In Revision 15 of the DCD, one of the reactor trip functions used for the "startup of an inactive reactor coolant pump at an incorrect temperature" event is the low-flow trip with P-8 interlock. In Revision 17, the applicant changed the P-8 interlock credited for this event to the P-10 interlock. As discussed above, the applicant deleted the P-8 interlock because the AP1000 is not licensed for N-1 loop operation. The power range nuclear power P-10 interlock, which has a lower setpoint than the P-8 interlock (10 percent and 48 percent power for P-10 and P-8, respectively), performs the same function of allowing reactor trip on low coolant flow or RCP high bearing water temperature in multiple loops that the P-8 interlock performs for single loop. Since the P-10 interlock has lower setpoint than the P-8 interlock, the replacement of the P-8 with the P-10 interlock is conservative because it would permit the reactor to be tripped at a lower power for this event. The staff also requested that the applicant confirm that safety analyses of transient events that take credit for the P-8 interlock have been reanalyzed with the P-10 interlock. The staff agrees with the applicant's response to RAI-SRP15.3.1-SRSB-01 that the replacement of the P-8 with the P-10 interlock will improve the results of the non-LOCA safety analysis, including for loss-of-flow events, at lower power since the reactor will now be tripped by a partial or completed loss of flow down to 10 percent power.

In DCD TS Table 3.3.1-1, the applicant replaced the P-8 interlock with the P-10 interlock for trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. The staff concludes that the change to DCD Table 15.0-6 to replace the P-8 interlock with the P-10 interlock is acceptable because it is consistent with the reactor trip logic and the TS.

15.1.0.6.4.2 Conclusion

The staff has reviewed the proposed changes to the DCD for the removal of the power range nuclear power reactor trip P-8 interlock and replacement with the P-10 interlock where applicable. The staff finds these changes conservative and acceptable because the P-10 interlock has a lower setpoint than the P-8 interlock, and the safety analyses of non-LOCA events credited with this interlock would continue to show compliance with GDC 10 and 15 without exceeding the specified acceptable fuel design limits and the RCS pressure boundary design limit.

15.1.0.6.5 Changes in Valve Opening Time Delay

In DCD Table 15.0-4a, the applicant changed the time delay for the automatic depressurization system (ADS) Stage 1 actuation on core makeup tank (CMT) low level signal from 20 seconds to 30 seconds for the control valve to begin to open. It also changed the ADS Stage 4 actuation on CMT low-low level signal from 30 seconds to 2 seconds for the squib valve to begin to open. In Table 15.0-4b, the applicant changed the closure time of the CVS makeup isolation valves from 10 to 30 seconds, and in Table 15.6.5-10, it also changed the control valve actuation and opening times for ADS Stages 1, 2, 3, and 4. It added a note to these tables, stating the following:

The valve stroke times reflect the design basis of the AP1000. The applicable DCD Chapter 15 accidents were evaluated for the design basis stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remains valid. The output provided for the analyses is representative of the transient phenomenon.

15.1.0.6.5.1 Evaluation

In RAI-SRP15.0-SRSB-04, the staff asked the applicant to clarify the relationship among the 2-second delay time of the ADS Stage 4 squib valves specified in Table 15.0-4a and the opening time delay sequences for Substage A and Substage B of ADS Stage 4 described in DCD Section 7.3.1.2.4, “Automatic Depressurization System Actuation,” and the ADS Stage 4 actuation time delay assumed in the safety analyses. In its response to RAI-SRP15.0-SRSB-04, the applicant provided the following explanation. In the small-break LOCA analysis in DCD Section 15.6.5.4B, ADS Stage 4 was actuated from the coincidence of CMT level less than the low-2 setpoint, low RCS pressure, and a preset time delay following an open signal initiation for the ADS Stage 3 depressurization valves. The 2-second time delay for ADS Stage 4 actuation on CMT low-2 level identified in Table 15.0-4a represents the accident analysis delay time assumed specifically for the squib valve to begin to open. The accident analysis assumes the 2-second signal processing delay after the CMT low-2 level is reached or the preset time delay after the ADS Stage 3 depressurization valve open signal is generated, whichever is later. This explanation clarifies that the ADS-4 opening time assumed in the safety analysis is for the ADS-4 Substage A depressurization valve opening time, and the Substage B depressurization valve will open after the Stage A valve with a delay time specified in DCD Table 15.6.5-10.

The applicant also identified in its response to RAI-SRP15.0-SRSB-04 the following inconsistencies in Revision 17 of the DCD, Chapters 7 and 15, which need to be revised:

- Section 7.3.1.2.4 states that ADS Stage 2 actuation is interlocked with ADS Stage 1, and ADS Stage 3 actuation is interlocked with ADS Stage 2. However, no such interlocks exist and, therefore, Section 7.3.1.2.4 requires revision.

- Table 15.0-4a indicates an ADS Stage 1 actuation on CMT low-level signal delay time of 30 seconds. This should be revised to 32 seconds, consistent with DCD Table 15.6.5-10, as it includes a 30-second programmable delay time with a 2-second signal proceeding delay.

- DCD Table 15.0.4b incorrectly lists Table 15.6.5-13 for the ADS valve opening times. The ADS valve opening times actually appear in Table 15.6.5-10.

- Table 15.6.5-10 should be modified to add two notes: (1) a description of the interlock of CMT low-2 level, as well as 128 seconds after the ADS Stage 3 actuation signal is generated for initiation of ADS-4 Substage A valves; and (2) that the valve opening time of ADS Stage 4 valves includes an “arm-fire” processing delay.

In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Because the PMS and equipment actuation delay times assumed in the safety analyses for various design-basis events may differ from the valve stroke time design-basis values listed in DCD Tables 15.0-4a, 15.0-4b, and 15.6.5-10, the staff asked the applicant to list the PMS and equipment with the delay times assumed in the Chapter 15 safety analyses that differ from the design-basis values specified in these tables and to evaluate the impact of the inconsistencies in delay time assumptions on the affected events. In its response to RAI-SRP15.0-SRSB-05, the applicant listed the PMS actuation functions and valves with different delay times. The list includes the ADS Stage 1 and Stage 4 actuations on CMT low level signals; the ADS Stage 1, 2, 3, and 4 control (depressurization) valves; and the CVS makeup isolation valves. The events analyzed with different delay times include boron dilution, inadvertent CVS actuation, loss of normal feedwater events analyzed with different CVS makeup isolation valve delay times, and the small-break LOCA and inadvertent operation of the ADS with different ADS valve opening time delays. The applicant evaluated the effects of the inconsistencies on the safety analyses of the affected events.

The design-basis stroke time for the CVS makeup isolation valves is 30 seconds, but the safety analyses for the inadvertent CVS actuation (DCD Section 15.5.2) and loss of normal feedwater (DCD Section 15.2.7) events assume a 12-second valve closure (which includes a 2-second microprocessor delay and 10-second closure time). Based on the CVS makeup flow rate assumed in the analysis, an additional 20 seconds in the makeup valve isolation time would increase the makeup flow into the RCS by about 0.425 cubic meters (m^3) (15 cubic feet (ft^3)). For these two events, there is sufficient available margin between the pressurizer volume and the maximum pressurizer water volume (see DCD Figures 15.5.2-5 and 15.2.7-6, respectively) to accommodate this water volume without changing the conclusion that the minimum DNB ratio remains above the design-limit value and the RCS pressure remains below 110 percent of the design value. For the boron dilution events, the applicant’s response to RAI-SRP9.3.6-SRSB-01 provides an evaluation. The longer makeup valve closure time results in no adverse effect for the boron dilution events occurring during Mode 1 and 2 operations because the purge volume of the CVS is not sufficient to return the reactor to criticality. For the events occurring during Mode 3, 4, and 5 operations, the safety analyses described in DCD Sections 15.4.6.2.2 through 15.4.6.2.4 assume a makeup valve closure time of 28 seconds, and there is sufficient margin to accommodate the 2-second delay in the valve closure time.

The only non-LOCA event affected by the ADS is an inadvertent operation of the two ADS Stage 1 trains event, since all other non-LOCA events do not model the ADS valves. The existing analysis described DCD Section 15.6.1 for the inadvertent operation of the ADS Stage 1 valve event assumed a 25-second stroke time, compared to the design-basis value of 40 seconds. As shown in DCD Figure 15.6.1-7, the minimum DNB ratio occurs around 22 seconds, which is before the ADS valves are fully open. Therefore, the longer valve stroke time has no effect on the analysis result and the existing analysis remains valid.

Since the primary role of the ADS is to mitigate a small-break LOCA, the staff asked the applicant to evaluate the effects of changes in the ADS valve stroke times on the small-break LOCA analysis. In response to RAI-SRP5.4.6-SRSB-01, the applicant evaluated the effects of the changes in the ADS Stage 1, 2, and 3 stroke times, as well as the ADS Stage 4 time delay change from 30 seconds to 2 seconds, on the small-break LOCA analysis described in DCD Section 15.6.5.4B, “Small-break LOCA Analyses.” The evaluation includes inadvertent ADS actuation, a 2-inch cold leg break, and a direct vessel injection (DVI) break. The results show minimal effects on the safety analysis and that the core remains covered in all cases; thus, the design continues to comply with the acceptance criteria for the ECCS specified in

10 CFR 50.46. Therefore, the staff concludes that the changes in the ADS valve actuation and stroke times specified in DCD Tables 15.0-4a and 15.6.5-10 are acceptable.

It should be noted that in Revision 17 of the DCD, Tier 1, Section 2.1.2, "Reactor Coolant System," and Section 2.3.2, "Chemical and Volume Control System," the applicant proposed to change the acceptance criteria for the opening times after receipt of a signal from the PMS of the remotely operated ADS Stage 1, 2, and 3 valves (RCS-V001A/B, RCS-V002A/B, and RCS-V003A/B) and the CVS makeup isolation valves (V090 and V091) specified in Tables 2.1.2-4 and 2.3.2-4, "Inspections, Tests, Analyses, and Acceptance Criteria," for the RCS and CVS, respectively. The staff finds that the revised values are acceptable because they are consistent with the values described in DCD Tier 2, Tables 15.0-4b and 15.6.5-10.

15.1.0.6.5.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes in DCD Tables 15.0.4a, 15.0.4b, and 15.6.5-10, regarding the delay times of the ADS valves and CVS makeup isolation valves are acceptable because they produce insignificant changes in the results of the analyses for the affected design-basis events, and the acceptance criteria in GDC 10, GDC 15, and 10 CFR 50.46 continue to be met.

15.1.0.6.6 Changes Pertaining to Steam Generator Tube Rupture Analysis

In Revision 17 of the DCD, the applicant made the following changes to Table 15.0-4a:

- Change the high-2 steam generator limiting setpoint from 100 percent to 95 percent of the narrow range level span.

- Add an entry for CMT actuation on pressurizer low-2 water level with a time delay of 2.0 seconds.

15.1.0.6.6.1 Evaluation

In its response to RAI-SRP15.0-SRSB-06, the applicant confirmed that it made these changes to clarify the DCD documentation in areas identified during the AP1000 DCD review process. The changes do not represent new assumptions or results and are consistent with the assumptions credited in the existing Chapter 15 safety analyses. The change of the high-2 steam generator limiting setpoint from 100 percent to 95 percent of the narrow range level span is consistent with the steam generator tube rupture analysis provided in DCD Section 15.6.3. Other events used 100 percent of the narrow range level span, but the steam generator tube rupture analysis provides the limiting setpoint. The addition of the CMT actuation on pressurizer low-2 water level with a time delay of 2 seconds is also consistent with the steam generator tube rupture analysis, which credits this signal. The staff, therefore, concludes that these changes are acceptable.

15.1.0.6.6.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes in DCD Table 15.0-4a pertaining to the steam generator low-2 setpoint and CMT actuation are acceptable because they are made merely for clarification of the DCD documentation and do not represent new assumptions in the steam generator tube rupture analysis.

15.1.0.8 Plant Systems and Components Available for the Mitigation of Accident Effects

15.1.0.8.1 Change to Chemical and Volume Control System Makeup Suction Isolation for Boron Dilution Mitigation

In DCD Table 15.0-6, “Plant Systems and Equipment Available for Transient and Accident Conditions,” the applicant changed the equipment credited for the “chemical and volume control system malfunction which results in a boron dilution” event from “low insertion limit annuciators” to “CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system.”

15.1.0.8.1.1 Evaluation

For the event of boron dilution resulting from CVS malfunction, the AP1000 design takes credit for the source range flux doubling signal (as discussed in Section 15.1.0.6.3 above), which will automatically isolate the unborated water source for boron dilution protection. Revision 17 of the DCD changes the CVS makeup valve realignment for boron dilution protection. The applicant revised DCD Section 9.3.6.4.5.1, “Boron Dilution Events,” to state that the CVS is designed to address a boron dilution accident by closing redundant safety-related valves, tripping the makeup pumps, or aligning the suction of the makeup pumps to the boric acid tank, or all three. It revised Section 9.3.6.3.7, “Chemical Volume and Control System Valves,” to state that these normally open, motor-operated makeup line containment isolation valves close on a source range flux doubling signal to terminate possible unplanned boron dilution events. The applicant also revised Section 7.3.1.2.14, “Boron Dilution Block,” to state the following:

> In the event of an excessive increasing rate of source range flux doubling signal, the block of boron dilution is accomplished by closing the chemical and volume control system makeup isolation valves and closing the makeup pump suction valves to the demineralized water storage tanks. This signal also provides a non-safety trip of the makeup pumps. These actions terminate the supply of potentially unborated water to the reactor coolant system as quickly as possible.

Therefore, the staff finds that the change in Table 15.0-6 regarding the use of CVS makeup pump suction isolation valves for the mitigation of boron dilution is consistent with the CVS design. As discussed in Section 15.2.4.6 of this report regarding boron dilution events occurring during Mode 3, 4, and 5 operations, this design change would terminate the boron dilution event sooner and has no safety significant effect on the consequence of the boron dilution events. Therefore, the staff concludes that this change is acceptable.

15.1.0.8.1.2 Conclusion

Based on the above evaluation, the staff concludes that the change in DCD Table 15.0-6, regarding the change to close the makeup line isolation valves for the termination of boron dilution events is acceptable because it reflects the CVS design change, has no significant effect on the consequence of boron dilution events, and the applicable GDC continue to be met.

15.1.0.8.2 Other Changes for Clarification

In Revision 17 of the DCD, the applicant made the following changes to Tables 15.0-6 and 15.0-8:

- Add the main steam isolation valves (MSIVs), startup feedwater isolation, and accumulators credited for the analyses of the inadvertent opening of a steam generator safety valve and steam system pipe failure events (DCD Sections 15.1.4 and 15.1.5).

- Add the steam generator safety valves for the analysis of the inadvertent operation of the CMT during power operation (DCD Section 15.5.1).

- Add the low steamline pressure ESF actuation functions for the analysis of the CVS malfunction that increases reactor coolant inventory (DCD Section 15.5.2).

- Add the low pressurizer level ESF actuation function for the analysis of the steam generator tube rupture (DCD Section 15.6.3).

- Add an entry to specify that the sample line isolation valves are credited for the failure of small lines carrying primary coolant outside containment (DCD Section 15.6.2).

- Revise the footnote to Table 15.0-8 pertaining to the MSIV backup valves from stating “moisture separator reheat steam supply control valve” to “moisture separator reheater 2nd stage steam isolation valves.”

15.1.0.8.2.1 Evaluation

In its response to RAI-SRP15.0-SRSB-06, the applicant confirmed that it made these changes merely to clarify the DCD documentation in areas identified during the AP1000 DCD review process. The changes are made to be consistent with the assumptions credited in the corresponding DCD sections in existing Chapter 15 safety analyses, and they do not represent new assumptions or results.

15.1.0.8.2.2 Conclusion

The staff has reviewed the above changes in DCD Tables 15.0-6 and 15.0-8. The staff concludes that these changes are acceptable because they are made merely for clarification of the DCD documentation and do not represent new assumptions or affect the results in the existing Chapter 15 safety analyses.

15.1.0.12 Component Failures

In Revision 17 of the DCD, the applicant proposed to delete the following paragraph in Section 15.0.12.1, pertaining to operator action error:

> A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety related equipment and does not include thought process errors that could potentially lead to common cause or multiple errors.

15.1.0.12.1 Evaluation

In Revision 2 of response to RAI-SRP15.0-SRSB-03 dated July 8, 2010, the applicant stated that while the Chapter 15 safety analyses do meet the intent of the statement in DCD Section 15.0.12.1, the statement was originally removed to prevent confusion about the operator action assumptions made for in the DCD safety analyses. There were no changes to the safety analysis assumptions as a result of the removal of this statement. Based on a review of the statement in DCD Section 15.0.12.1, it is determined that the deleted statement will be returned to the DCD, with the punctuation of the statement updated as follows:

> A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure; the error is limited to manipulation of safety related equipment and does not include thought process errors that could potentially lead to common cause or multiple errors.

Therefore, except for the editorial change in the above statement, no change is made to the certified DCD Revision 15 pertaining to operator actions. The staff finds it acceptable.

15.1.0.12.2 Conclusion

As discussed above, the applicant has determined to retain the statement in the certified DCD Revision 15 pertaining to operator actions, except for an editorial change. This is acceptable. In a subsequent revision to the AP1000 DCD, the applicant incorporated this change in the DCD text, which resolves this issue.

15.2 Transients and Accident Analysis

15.2.2 Decrease in Heat Removal by the Secondary System (DCD Tier 2, Section 15.2)

15.2.2.6 Loss of Alternating Current Power to the Plant Auxiliaries (DCD Tier 2, Section 15.2.6)

In Revision 17 of the DCD, Section 15.2.6.2.1, "Method of Analysis," the applicant added the statement that "the main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative." In addition, it deleted the statement from the assumption used in the analysis that "conservative PRHR heat exchanger heat transfer coefficients (low) associated with the low flow rate caused by the reactor coolant pump trip are assumed."

15.2.2.6.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.2.6, "Loss of Power to the Plant Auxiliaries," describes the analysis of the loss of power to the plant auxiliaries caused by a complete loss of offsite grid, accompanied by a turbine-generator trip. Section 15.2.2.6 of NUREG-1793, "Final Safety Evaluation Report

Related to Certification of the AP1000 Standard Design," described the staff's evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1-percent power uncertainty will be performed through COL Item 15.0-1. However, the changes have no effect on the analysis of the loss of alternating current power to the plant auxiliaries because the analysis assumes 2-percent power uncertainty. In addition, the applicant deleted the conservative PRHR heat exchanger heat transfer coefficient to provide more accurate information consistent with the existing analysis of the event. The staff concludes that these changes have no effect on the existing analysis of the event, which continues to show compliance with the relevant acceptance criteria; therefore, the changes are acceptable.

15.2.2.6.2 Conclusion

Based on the above evaluation, the staff concludes that the revisions proposed by the applicant to the DCD, Section 15.2.6.2.1, do not represent new assumptions and have no effects on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.2.7 Loss of Normal Feedwater Flow (DCD Tier 2, Section 15.2.7)

Revision 17 of the DCD added the following statement to Section 15.2.7.2.1, "Method of Analysis," regarding the assumptions used in the analysis: "The main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2-percent power uncertainty is conservative." The applicant also made editorial changes to Table 15.2-1 to correct the time sequence of the loss of normal feedwater flow event; these changes have no effect on the analysis.

15.2.2.7.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.2.7, "Loss of Normal Feedwater Flow," describes the analysis of the loss of normal feedwater flow. Section 15.2.2.7 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1-percent power uncertainty will be performed through COL Item 15.0-1. However, the change has no effect on the analysis of the loss of normal feedwater flow because the analysis assumes 2 percent power uncertainty. Therefore, the staff concludes that this change is acceptable.

15.2.2.7.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.2.7, do not represent new assumptions and have no effects on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.2.8 Feedwater System Pipe Break (DCD Tier 2, Section 15.2.8)

Revision 17 of the DCD changes Section 15.2.8.1, "Identification of Causes and Accident Description," to include the high-3 pressurizer water level as a condition for a reactor trip. The applicant also added the statement, "Method of Analysis," that "the main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2 percent power uncertainty is conservative," to Section 15.2.8.2.1.

15.2.2.8.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.2.8, "Feedwater System Pipe Break," describes the analysis of feedwater system pipe break. Section 15.2.2.8 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

In its response to RAI-SRP15.0-SRSB-06, the applicant stated that it added the high-3 pressurizer water level as a reactor trip actuation condition in order to provide more accurate information in the DCD to be consistent with the existing safety analysis assumption. The staff finds the change acceptable since it does not represent a new assumption and has no effect on the analysis result.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1 percent power uncertainty will be performed through COL Item 15.0-1. However, the change has no effect on the analysis of the feedwater system pipe break because the analysis assumes 2 percent power uncertainty. Therefore, the staff concludes that this change is acceptable.

15.2.2.8.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.2.8, do not represent new assumptions and have no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.3 Decrease in Reactor Coolant System Flow Rate (DCD Tier 2, Section 15.3)

15.2.3.1 Partial Loss of Forced Reactor Coolant Flow (DCD Tier 2, Section 15.3.1)

In DCD, Section 15.3.1, the applicant proposed to change the power range nuclear power reactor trip permissive from the P-8 interlock to the P-10 interlock for protection against the partial loss of forced reactor coolant flow event.

15.2.3.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.3.1 describes the safety analysis of the partial loss of forced reactor coolant flow event. Section 15.2.3.1 of NUREG-1793 described the staff evaluation of this event. Therefore, the evaluation in this document addresses the staff evaluation of the proposed change described in Revision 17 of the DCD.

DCD Section 15.3.1.1, “Identification of Causes and Accident Description,” describes the cause and progression of the partial loss of forced reactor coolant flow event (i.e., trip of one RCP). The low primary coolant flow reactor trip signal provides protection against this event. Revision 15 of the DCD stated that above permissive P8, low flow in either hot leg actuates a reactor trip, and between approximately 10 percent power (permissive P10) and the power level corresponding to P8, low flow in both hot legs actuates a reactor trip. Revision 17 of the DCD revises this to state that above permissive P10, low flow in either hot leg actuates a reactor trip.

As discussed in Section 15.1.0.6.4 of this report, Revision 17 of the DCD deletes the P-8 interlock because the P-8 interlock, which permits reactor trip on low flow or RCP high bearing temperature in a single loop, is not needed for the AP1000 since it is not licensed for N-1 loop operation. In Revision 17 of the DCD, the applicant deleted the P-8 interlock from Table 7.2-3, “Reactor Trip Permissives and Interlocks.” Therefore, the P-8 permissive interlock is changed to P-10 interlock for the low-flow trip function. The P-10 interlock also replaces the P-8 interlock in TS Table 3.3.1-1, trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. Therefore, the change to replace the P-8 interlock with the P-10 interlock is consistent with the reactor trip logic and TS. As stated in its response to RAI-SRP15.3.1-SRSB-01, the change from the P-8 interlock to the P-10 interlock will improve the results of the loss-of-flow events at lower powers since the reactor will now be tripped at the P-10 setpoint of 10-percent power, compared to the P-8 setpoint of 48 percent power. As a result of replacing the P-8 permissive with P-10, the reactor trip with low flow in both hot legs when the power level is between approximately 10 percent (permissive P-10) and that corresponding to permissive P8 (48 percent) is not necessary. The staff concludes that the change to DCD Section 15.3.1.1 is acceptable.

15.2.3.1.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.3.1, to replace the P-8 interlock with the P-10 permissive interlock are conservative relative to the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.3.2 Complete Loss of Forced Reactor Coolant Flow (DCD Tier 2, Section 15.3.2)

In DCD, Section 15.3.2, the applicant proposed to change the power range nuclear power reactor trip permissive from the P-8 interlock to the P-10 interlock for protection against the complete loss of forced reactor coolant flow event.

15.2.3.2.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

DCD Section 15.3.2, “Complete Loss of Forced Reactor Coolant,” describes the safety analysis of the complete loss of forced reactor coolant flow event. Section 15.2.3.2 of NUREG-1793 described the staff evaluation of this event. Therefore, the evaluation in this document addresses the staff evaluation of the proposed change described in Revision 17 of the DCD.

DCD Section 15.3.2.1, “Identification of Causes and Accident Description,” describes the cause and progression of the complete loss of forced reactor coolant flow event (i.e., trip of all four RCPs). The low primary coolant flow reactor trip signal provides protection against this event. Revision 15 of the DCD stated that above permissive P8, low flow in either hot leg actuates a reactor trip, and between approximately 10 percent power (permissive P10) and the power level corresponding to P8, low flow in both hot legs actuates a reactor trip. Revision 17 of the DCD revises this to state that above permissive P10, low flow in either hot leg actuates a reactor trip.

Section 15.2.3.1 of this report discusses the staff evaluation that concludes that the replacement of the P8 permissive interlock with the P10 interlock for the low reactor coolant flow is acceptable. This same conclusion applies to the complete loss of forced reactor coolant flow. Therefore, the staff concludes that the change to DCD Section 15.3.2.1 is acceptable.

15.2.3.2.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the DCD, Section 15.3.2, to replace the P-8 interlock with the P-10 permissive interlock are conservative relative to the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

15.2.4 Reactivity and Power Distribution Anomalies (DCD Tier 2, Section 15.4)

15.2.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition (DCD Tier 2, Section 15.4.1)

In DCD, Section 15.4.1.1, the applicant proposed to change the high nuclear flux rate reactor trip. This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above a preset setpoint. Previously, this trip function could be manually bypassed after the coincident two out of four nuclear power range channels were manually reset. The applicant proposed to no longer allow the manual bypass of the trip function after the manual reset of the coincident two out of four nuclear power range channels.

15.2.4.1.1 Evaluation

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

GDC 13, "Instrumentation and Control," requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 17, "Electric Power Systems," requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components (SSCs) important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that specified acceptable fuel design limits and design conditions of the RCPS are not exceeded as a result of AOOs. Meeting GDC 17 ensures that the fuel cladding integrity is not challenged during an uncontrolled control rod assembly withdrawal in conjunction with a loss of onsite or offsite power.

GDC 20, "Protection System Functions," requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

The staff has reviewed the proposed change by the applicant to no longer allow the manual bypass of the high nuclear flux rate reactor trip function after the manual reset of the coincident two out of four nuclear power range channels. Since the proposed change would no longer allow manual bypassing of the trip function, the proposed change has no effect on the analysis results for the control rod withdrawal from subcritical transient. Therefore, the staff concludes that this proposed change is acceptable.

15.2.4.1.2 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable because it continues to meet the requirements of GDC 10, GDC 13, GDC 17, GDC 20, and GDC 25.

15.2.4.3 Rod Cluster Control Assembly Misalignment (DCD Tier 2, Section 15.4.3)

In DCD, Section 15.4.3.1, the applicant proposed to delete a sentence pertaining to a specific operator action upon the inoperability of the rod deviation alarm to be consistent with the TS. The resolution of the rod position indicator channel is ±5 percent of span (±7.5 inches). A deviation of any rod cluster control assembly (RCCA) from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions greater than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. The applicant proposed to delete the sentence from the application that states, “If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specification.” However, the following paragraph in the application states that if one or more of the rod position indicator channels are out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs and the operator also takes action as required by the TS.

15.2.4.3.1 Evaluation

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

GDC 25 requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

In its response to RAI-SRP15.4.3-SRSB-01, the applicant stated that it revised the sentence, "If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specifications," in Revision 16 of the DCD, Section 15.4.3.1, to be consistent with TS 3.1.4 and 3.1.7. The applicant also stated that neither of these TS or any other TS require the rod deviation alarm to be operable. The applicant also added that Revision 3 of the DCD revised TS 3.1.4 and 3.1.7 to be consistent with NUREG-1431, "Standard Technical Specifications—Westinghouse Plants," Revision 2, issued April 2001. NUREG-1431, Revision 2, removes the rod deviation alarm since it serves as indication only and does not directly relate to the limiting conditions for operation. This is documented in TSTF-110, "Delete SR frequencies based on inoperable alarms," Revision 2. AP1000 TS 3.1.4 does require that all shutdown and control rods shall be operable and that individual indicated rod positions shall be within 12 steps of their group step counter demand position. Surveillance Requirement (SR) 3.1.4.1 requires that individual rod positions are verified within alignment limit every 12 hours. The applicant stated that performing this verification every 12 hours provides a history that allows the operator to detect that a rod is beginning to deviate from its expected position. In addition, the specified frequency takes into account other rod position information that is continuously available to the operator in the main control room (MCR) so that during actual rod motion, deviations can immediately be detected. According to the applicant, the digital rod position indication system and the bank demand position indication system make rod position information continuously available to the operator in the MCR. TS 3.1.7 requires the digital rod position indication system and the bank demand position indication system to be operable. The digital rod control system maintains a count of steps taken by each rod group and, based on this information, a digital readout of the demanded bank position is provided and the demanded and measured rod position signals are displayed in the MCR. An alarm is generated whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. The applicant verifies that the alarm is set with appropriate allowance for instrument error and within sufficiently narrow limits to prevent exceeding core design hot channel factors.

The staff has reviewed the proposed change by the applicant to delete the need for operator action if the rod deviation alarm is not operable. Based on its review, the staff finds that this revision does not affect the safety analysis of the RCCA misalignment events described in DCD Section 15.4.3, and the design continues to comply with the relevant requirements. Therefore, the staff concludes that this proposed change is acceptable.

15.2.4.3.2 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable because it continues to meet the requirements of GDC 10, 13, 20, and 25.

15.2.4.6 Chemical and Volume Control System Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant (DCD Tier 2, Section 15.4.6)

The applicant proposed to revise Section 15.4.6 in the areas related to the CVS design modifications associated with the mitigation of boron dilution events. These modifications include the alignment of the makeup pump suction, and flux doubling boron dilution block and CVS isolation pertaining to boron dilution events occurring during Mode 3, 4, and 5 operations. In addition, the applicant made changes regarding the dilution flow rates, RCS mixing volume, and critical and shutdown boron concentrations for Mode 3, 4, and 5 operations.

15.2.4.6.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs; and GDC 13 and GDC 26, "Reactivity Control System Redundancy and Capability," pertaining to appropriate instrumentation and reactivity control system redundancy and capability.

DCD Section 15.4.6 describes the safety analysis for the boron dilution event. Section 15.2.4.6 of NUREG-1793 described the staff evaluation of this event for compliance with relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

In Section 15.4.6.1, "Identification of Causes and Accident Description," the applicant proposed to modify the description of boron dilution events and the general mitigation method to be consistent with the CVS design changes for boron dilution mitigation described in DCD Section 9.3.6, "Chemical Volume and Control System." The existing design currently realigns the makeup pump suction from the demineralized water tank to the boric acid tank to terminate the potential boron dilution, and to begin to reborate the RCS to restore shutdown margin. These actions would initially cause the boron dilution to continue because the volume of water in the makeup line path would still be unborated until borated water from the boric acid tank began to reach the RCS. The function was changed to close the makeup line isolation valves (as well as the demineralized water isolation valves) or trip the makeup pumps to terminate the event as soon as possible. Long-term recovery from the event would then be accomplished using either a different flowpath with a smaller unpurged volume or by using the makeup line after purging most of the unborated water in it. The applicant also proposed several text changes along with the logic changes that are required to implement this modication.

The staff reviewed the applicant's proposed change related to the realignment from the demineralized water tank to the boric acid tank as well as isolation of the makeup flow to the RCS for the termination of boron dilution events in Modes 3, 4, and 5. Since the proposed change would terminate the boron dilution event sooner and has no safety-significant effect on the transient, the staff finds this change acceptable. The text changes associated with the realignment are, therefore, acceptable.

In DCD, Section 15.4.6.2, the applicant proposed to delete text that states that in the event of an inadvertent boron dilution transient during Mode 3, 4, and 5 operations, the source range nuclear instrumentation detects "an increase of 60 percent of" the neutron flux and replace it with generic text reading “a sufficiently large increase in” the neutron flux. In its response to RAI-SRP-15.4.6-SRSB-01, the applicant explained that in Revision 15 of the DCD, the use of the phrase “an increase of 60 percent” reflected the 1.6 multiplier (over 50 minutes) setpoint reported in DCD Table 15.0-4a. In Revision 16 of the DCD, and supplemented by TR-80, the applicant proposed to change the multiplier from 1.6 to 2.2 (which would now be an increase of greater than 60 percent). The applicant stated that it reanalyzed the event and verified that an increase in the nominal setpoint from 1.6 to 2.2 demonstrates acceptable results and would reduce the likelihood of an inadvertent actuation of the boron dilution protection functions during normal operation.

The applicant changed Revision 16 of the DCD, Table 15.0-4a, to reflect the proposed 2.2 multiplier. The applicant stated that it decided to remove the text in DCD Section 15.4.6 referencing a specific flux increase value in order to simplify any potential future revisions to the text in the event of a subsequent change in the safety analysis setpoint. By reporting the actual numerical value in DCD Table 15.0-4a, the document completely defines the magnitude of the flux increase modeled in the safety analysis; therefore, the applicant stated that there is no need to repeat that same information in multiple locations within the text. The staff has reviewed the applicant’s justification for replacing the text referencing “an increase of 60 percent” with generic text reading “a sufficiently large increase.” Since DCD Table 15.0-4a will provide a specific value to quantify the phrase, “a sufficiently large increase,” this change is acceptable.

However, in its responses to both RAI-SRP-15.4.6-SRSB-01 and RAI-SRP15.0-SRSB-01, the applicant proposed to increase the multiplier again from 2.2 to 3.0. In its response to RAI-SRP15.0-SRSB-01, the applicant stated that the safety analysis cases reported in DCD Section 15.4.6 support a setpoint of 3.0 over 50 minutes to mitigate the boron dilution event in Modes 3, 4, and 5. However, DCD Section 15.4.6 and TR-80 only discuss the increase from 1.6 to 2.2 and do not mention an increase to 3.0. The applicant submitted Revision 17 of the DCD, Table 15.0-4a, which includes the new proposed flux rate setpoint of 3.0, but the staff asked the applicant to provide additional information regarding the 3.0 multiplier. In response to RAI-SRP15.4.6-SRSB-02, the applicant confirmed that the boron dilution analyses performed for Modes 3, 4, and 5 documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes, and analysis results for all cases are acceptable. The applicant also justified why the 36-percent allowance between the safety analysis and TS nominal setpoints of 3.0 and 2.2 over 50 minutes is sufficient to bound the actual value determined when final plant design inputs are available. The staff reviewed the applicant’s response confirming that the boron dilution analyses documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes, and it finds this change acceptable. The staff also reviewed the applicant’s justification that the 36-percent allowance between the safety analysis and TS nominal setpoints of 3.0 and 2.2 is sufficient to bound the actual value determined when the final plant design inputs are available, and it finds this explanation acceptable.

Additionally, in DCD, Section 15.4.6.2, the applicant proposed changes to the dilution flowrate, RCS water volume, critical and shutdown boron concentrations, and automatic protective actions initiation time for Mode 3, 4, and 5 operations. The applicant proposed these changes in order to be consistent with the TS, DCD Section 9.3.6, and the assumed conditions for the inadvertent boron dilution event. In its response to RAI-SRP15.4.6-SRSB-03, the applicant provided additional explanation for the changes in the RCS water volumes for different modes of operation. The applicant stated that it recalculated the RCS water volumes using the latest geometric data available, taking into consideration the design changes made up to this point.

These design change refinements result in a change of less than 5 percent for Mode 3 and of less than 8 percent for Mode 5. The Mode 4 RCS volume is the same volume as was assumed in the Mode 3 boron dilution calculation and no credit is taken for the upper head volume in the assumed active mixing volume in the Mode 3, 4, and 5 calculations. The staff reviewed the applicant's rationale for the changes in RCS water volumes for different modes of operation and concludes that these changes are acceptable.

15.2.4.6.2 Conclusion

The staff reviewed the proposed changes described in DCD, Section 15.4.6, regarding the CVS design modifications associated with the mitigation of boron dilution events. Based on the above evaluation, the staff concludes that the AP1000 design changes are acceptable because the analyses of boron dilution events continue to meet the requirements of GDC 10, 13, 15, and 26.

15.2.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents (DCD Tier 2, Section 15.4.8)

DCD Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," describes the RCCA ejection events resulting from mechanical failure of control rod mechanism pressure housing. For assemblies initially inserted, the consequences include a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to render this accident extremely unlikely, the applicant has provided its analysis of the consequences of such an event. The applicant has considered plant systems and equipment, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it determined that no single active failure in these systems or equipment adversely affects the consequences of the events.

Revision 17 of the DCD removes the longitudinal and circumferential failures described in Sections 15.4.8.1.1.5 and 15.4.8.1.1.6, respectively. These changes are supported by APP-GW-GLE-016, "Impact of In-core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package (IHP)," Revision 0.

15.2.4.8.1 Evaluation

Revision 17 of the DCD removes the failure mechanisms outlined in Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 because of a design change to the upper internals. TR APP-GW-GLE-016 provides a technical description of these changes, but the technical justification for removing the failure mechanisms was unclear.

The staff sent RAI-SRP15.4.8-SRSB-01 to the applicant to ask for more information as to why the failure mechanisms previously covered in DCD Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 are no longer considered. In its response to RAI-SRP15.4.8-01, the applicant stated that it removed the failure mechanisms from Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 because Section 3.9.4.1.1 covers RCCA failure mechanisms, and these methods were no longer considered credible. The RAI response further explains that the hypothetical failure of a RCCA housing described in DCD Section 15.4.8 is a rapid positive reactivity insertion independent of the specific failure mechanism; therefore, DCD Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 are not needed.

The staff reviewed the RAI response and agrees that as long as the analysis continues to evaluate a rapid positive reactivity insertion resulting from an ejected rod, the requirements of

GDC 28, “Reactivity Limits” (as detailed in regulatory guide (RG) 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” issued May 1974, and NUREG-0800 Sections 15.4.8 and 4.2), are met. Therefore, the staff concludes that the proposed change is acceptable.

15.2.4.8.2 Staff Position Related to the Revision of NUREG-0800 Section 15.4.8

In Section 15.2.4.8 of NUREG-1793, the staff provided a safety evaluation of the RCCA ejection analysis in accordance with NUREG-0800 Section 15.4.8, “Spectrum of Rod Ejection Accidents (PWR),” Revision 2, issued July 1981. The AP1000 analysis results of the RCCA ejection events initiated at hot full power and at hot zero power demonstrated that the calculated values of the hot spot radially averaged fuel enthalpy are well within the acceptance criterion of 280 calories per gram specified in NUREG-0800 Section 15.4.8, Revision 2.

After the AP1000 design certification (DC) rulemaking, the NRC revised NUREG-0800 Section 15.4.8. Revision 3 of NUREG-0800 Section 15.4.8, issued in March 2007, specifies that the number of failed fuel rods used in the radiological evaluation for the rod ejection events be calculated considering the failure mechanisms addressed in NUREG-0800 Section 4.2. Appendix B, “Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents,” to NUREG-0800 Section 4.2, Revision 3, specifies that, for the reactivity-initiated accidents such as rod ejection accidents in pressurized-water reactors (PWRs), the total number of fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing either: (1) the high cladding temperature failure criteria specified; or (2) the pellet-cladding mechanical interaction cladding failure criteria specified.

In 10 CFR 52.47(a)(9), “Contents of applications; technical information,” the NRC specifies that for applications for light-water reactor (LWR) nuclear power plants, the technical information in the application shall include an evaluation of the standard plant design against the NUREG-0800 revision in effect 6 months before the docket date of the application. In addition, 10 CFR 52.63(a)(1), “Finality of standard design certifications,” specifies that notwithstanding any provision in 10 CFR 50.109, “Backfitting,” while a standard DC rule is in effect under 10 CFR 52.55, “Duration of certification,” or 10 CFR 52.61, “Duration of renewal,” the Commission may not modify, rescind, or impose new requirements on the certified information, whether on its own motion or in response to a petition from any person, unless the Commission determines in a rulemaking that the change meets one of seven criteria. In 10 CFR 52.79(a)(41), “Contents of applications; technical information in final safety analysis report,” the NRC specifies that for applications for LWR nuclear power plant COLs, the technical information in the final safety analysis report (FSAR) shall include an evaluation of the facility against the NUREG-0800 revision in effect 6 months before the docket date of the application. In addition, 10 CFR 52.98(c)(1), “Finality of combined licenses; information requests,” specifies that if the COL references a certified design, then changes to or departures from information within the scope of the referenced DC rule are subject to the applicable change processes in that rule. Section VIII, “Processes for Changes and Departures,” in Appendix C, “Design Certification Rule for the AP600 Design,” and Appendix D, “Design Certification Rule for the AP1000 Design,” to 10 CFR Part 52, “Licenses, certifications, and approvals for nuclear power plants,” specifies the processes for changes and departures from the Tier 1, Tier 2, and Tier 2* information, respectively, in the certified design. For example, Section VIII.B.6 specifies that an applicant or licensee who references this appendix may not depart from Tier 2* information without NRC approval.

Accordingly, the staff determined that the AP600 and AP1000 certified standard designs, which the NRC certified before 2007, are not required to comply with Revision 3 of NUREG-0800 Sections 15.4.8 and 4.2, issued March 2007. For COL applicants or licensees who reference the AP1000 or AP600 certified designs, the staff will review any change or departure from the certified design that requires prior NRC approval as specified in Section VIII of Appendices C and D to 10 CFR Part 52, respectively.

The staff will evaluate the reactivity-initiated accidents such as rod ejection accidents based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapter 4, “Reactor,” or Chapter 15, “Transient and Accident Analysis.”

15.2.4.8.3 Conclusion

The staff concludes that the RCCA ejection analysis meets the acceptance criteria specified in NUREG-0800 Section 15.4.8, Revision 2, which were in effect for the AP1000 DC application. Therefore, the RCCA ejection analysis in the DCD is acceptable and can be used by COL applicants referencing the AP1000 standard design. However, the staff will evaluate the RCCA ejection accident based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapters 4 or 15.

15.2.5 Increase in Reactor Coolant System Inventory (DCD Tier 2, Section 15.5)

13.2.5.1 Inadvertent Operation of the Core Makeup Tanks during Power Operation (DCD Tier 2, Section 15.5.1)

The applicant proposed to revise Section 15.5.1.2 by deleting the sentence, “No single active failure in any of these systems or equipment adversely affects the consequences of the accident.”

15.2.5.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the NUREG-0800. These acceptance criteria include GDC 10 and GDX 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

In Section 15.5.1.2, the applicant proposed to delete the following sentence:

> No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

As stated in DCD Section 15.5.1.2, regarding the PMS actuations for the mitigation of the event, the PRHR heat exchanger removes the core decay heat. The worst single failure assumed is the failure of one of the two parallel isolation valves in the outlet line of the PRHR heat exchanger to open. Therefore, the single active failure is assumed in the safety analysis. In its response to RAI-SRP15.5.1-SRSB-02, the applicant clarified that it deleted this sentence to remove contradictory information, and the change has no effect on the existing safety analysis assumptions or methodology. Therefore, the staff concludes that this change is acceptable.

15.2.5.1.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to DCD, Section 15.5.1, to delete a contradicting statement are acceptable because there is no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15.

15.2.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (DCD Tier 2, Section 15.5.2)

The applicant proposed to revise Section 15.5.2, “Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory,” by deleting the sentence, “No single active failure in any of these systems or equipment adversely affects the consequences of the accident.”

15.2.5.2.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

In Section 15.5.2.2, the applicant proposed to delete the following sentence:

> No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

As stated in DCD Section 15.5.2.2, regarding the PMS actuations for the mitigation of the event, the PRHR heat exchanger removes the core decay heat. The worst single failure assumed is the failure of one of the two parallel isolation valves in the outlet line of the PRHR heat exchanger to open. Therefore, the single active failure is assumed in the safety analysis. In response to RAI-SRP15.5.1-SRSB-02, the applicant clarified that it deleted this sentence to remove contradictory information, and the change has no effect on the existing safety analysis assumptions or methodology. Therefore, the staff concludes that the change is acceptable.

15.2.5.2.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to DCD, Section 15.5.2, to delete a contradicting statement to be acceptable because there is no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and GDC 15.

15.2.6 Decrease in Reactor Coolant System Inventory (DCD Tier 2, Section 15.6, Excluding Section 15.6.5)

15.2.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the Automatic Depressurization System

The applicant proposed to revise Section 15.6.1, “Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS,” to change the motor-operated valve stroke times for the ADS valves for Stages 1–3.

15.2.6.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the NUREG-0800. These acceptance criteria include GDC 10 and GDC 15, which require that the specified acceptable fuel design limits and the design conditions of the RCPB, respectively, are not exceeded during any conditions of normal operation, including AOOs.

In Section 15.6.1.1, “Identification of Causes and Accident Description,” the applicant proposed the following changes to the motor-operated valve stroke times for ADS valves during this event:

- Change the ADS Stage 1 design opening time from 25 seconds to 40 seconds.
- Change the ADS Stage 2 and 3 design opening times from 70 seconds to 100 seconds.

The applicant also proposed to add the following paragraph to clarify the effects of the proposed times above on the analysis results:

> The valve stroke times shown in Chapter 15 tables (input/assumptions) reflect the design basis of the AP1000. The accidents addressed in this section were evaluated for these design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided in this section for the analyses is representative of the transient phenomenon.

The staff reviewed DCD, Section 15.6.1, and concludes that the proposed changes provide additional clarification and do not have a safety significant effect on the existing safety analysis of the event. Therefore, the proposed changes are acceptable.

15.2.6.1.2 Conclusion

Based on the above evaluation, the staff finds that the addition of the above paragraph in DCD, Section 15.6.1, only provides clarification and does not have a significant effect on the existing safety analysis. Therefore, the staff concludes that the proposed change is acceptable because the relevant requirements of GDC 10 and GDC 15 continue to be met.

15.2.6.5 Loss-of-Coolant Accident (DCD Tier 2, Section 15.6.5)

15.2.6.5.2 Large Breaks (DCD Tier 2, Section 15.6.5.4A)

In Revision 17 of the DCD, the applicant extensively revised Section 15.6.5.4A, “Large-Break LOCA Analysis Methodology and Results.” In Revision 15 of the DCD, the applicant performed the best estimate large-break loss-of-coolant accident (BELOCA) analysis using the WCOBRA/TRAC code to calculate thermal-hydraulic transients in the RCS during a postulated large-break LOCA, and it used the HOTSPOT program to calculate the effects of local models on the calculated peak cladding temperature (PCT). The uncertainties associated with the plant input parameters and states, such as initial fluid conditions in the reactor core system and the ECCS boundary conditions, were treated with a response surface method described in Westinghouse Commercial Atomic Power (WCAP)-12945-P-A, “Code Qualification Document [CQD] for Best Estimate LOCA Analysis,” Revision 2, issued 1998. In Revision 17 of the DCD, the applicant continued to use the WCOBRA/TRAC and HOTSPOT computer codes for the BELOCA thermal-hydraulic and hot fuel rod analyses. However, it used the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the statistical treatment of uncertainties, replacing the existing CQD response surface method. The NRC has reviewed and approved ASTRUM, documented in WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method,” issued 2005. Consistent with the CQD methodology, ASTRUM follows the steps of the Code Scaling, Applicability, and Uncertainty methodology. ASTRUM differs from the CQD methodology primarily in the statistical technique used for uncertainty treatment. ASTRUM uses a nonparametric statistical technique applied directly to a random sample of outputs, for example, the PCT, the maximum local oxidation (MLO), and the core-wide oxidation (CWO). These sample outputs are computed by applying Monte Carlo sampling of the inputs to the WCOBRA/TRAC and HOTSPOT calculations. The uncertainties and biases remain the same as in the CQD methodology.

In support of the revised BELOCA analysis in Revision 17 of the DCD, Section 15.6.5.4A, the applicant submitted APP-GW-GLE-026, “Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis,” Revision 1, issued 2009, which provides a detailed description of the revised analysis.

The staff used NUREG-0800 Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” and RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” issued May 1989, to guide its evaluation of this revised BELOCA analysis.

15.2.6.5.2.1 AP1000 BELOCA Analysis Code Applicability Evaluation

The BELOCA analysis uses the WCOBRA/TRAC code to calculate the effects of initial conditions, power distributions, and global models, and it uses the HOTSPOT code to calculate the effects of local models. The WCOBRA/TRAC code, described in WCAP-12945-P-A, is Westinghouse’s best estimate thermal-hydraulic computer code to evaluate the RCS response to a postulated large-break LOCA. Westinghouse developed the code consistent with the guidance provided in RG 1.157 to calculate thermal-hydraulic conditions in the RCS during blowdown and reflood of a LOCA. The code includes the features needed to satisfy the requirements of 10 CFR 50.46(a)(1)(i). The BELOCA analysis of the AP600 standard design used the WCOBRA/TRAC code. Section 21.6.3 of NUREG-1512, “Final Safety Evaluation Report Related to Certification of the AP600 Standard Design,” described the staff’s review of the application of WCOBRA/TRAC for the AP600 application. Revision 15 of the DCD used the "2000 Formulation" of WCOBRA/TRAC referenced in WCAP-15644-P, "AP1000 Code Applicability Report," Revision 2, issued 2004. This is the version approved in the analysis of the AP600 site safety analysis report, with the subsequent discretionary and nondiscretionary changes that Westinghouse reported to the NRC in 1998, 1999, and 2000, as presented in Appendix A to WCAP-15644-P, Revision 2. Section 21.6.3 of NUREG-1793 described the staff evaluation of WCOBRA/TRAC applicability to the AP1000 BELOCA.

The approved WCOBRA/TRAC code version used in the ASTRUM evaluation model described in WCAP-16009-P-A is WCOBRA/TRAC Mod 7, Revision 6, which is different from the 2000 Formulation of WCOBRA/TRAC used in Revision 15 of the DCD. The WCAP-16009-P-A code version includes the discretionary and nondiscretionary changes reported through the 10 CFR 50.46 reporting process.

The HOTSPOT program is a one-dimensional conduction code that models a portion of a fuel rod at the PCT or burst location and takes into account fuel relocation following a burst during a large-break LOCA. In the axial node where fuel rod burst is predicted to occur, the fuel relocation model in HOTSPOT is used to account for the likelihood that additional fuel-pellet fragments from above that elevation may settle into the burst region. Sections 25.4.2.3 and 25.4.2.4 of WCAP-12945-P-A describe the HOTSPOT model, application, and assessment. HOTSPOT uses a simple physical model that allows the effects of uncertainties to be calculated directly by running the model with parameter values that vary randomly according to specified distributions. The parameter uncertainties for the local models consider the power rates, fuel and cladding properties, metal-water reaction rates, and heat transfer coefficients.

15.2.6.5.2.1.1 WCOBRA/TRAC and HOTSPOT Code Modifications

Revision 17 of the DCD uses WCOBRA/TRAC (M7AR7_AP) and HOTSPOT (6.1) as the current code versions for the AP1000 revised BELOCA analysis. These versions differ from the earlier versions used in Revision 15 of the DCD and WCAP-16009-P-A because of additional code modifications identified through 10 CFR 50.46 reporting. For completeness, Appendix A to APP-GW-GLE-026 identifies a total of 35 changes through 10 CFR 50.46 reporting since 1998.

Appendix B to WCAP-16009-P-A identified 19 of the 35 changes and formed the bases for the acceptance of WCOBRA/TRAC Mod 7A, Revision 6, for BELOCA analyses. The applicant evaluated each of the changes and concluded that each error and its correction, or discretionary change did not have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties. The NRC evaluated these in its safety evaluation report for WCAP-16009-P-A. The agency concluded that these corrections were reasonable and effectual and found them to be acceptable. The remaining 16 changes were incorporated into WCOBRA/TRAC (M7AR7_AP) and HOTSPOT 6.1, the current code versions used for AP1000 BELOCA analyses.

Of the 16 changes, 10 are discretionary changes involving either: (1) enhanced input/output, corrections to output edits, or improvement in the automation of running the code cases; or (2) corrections to errors involving an option, model, code, or a code application that was not used for experiment simulations. The applicant evaluated each of these changes and concluded that each discretionary change did not have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties.

One of the nondiscretionary changes concerned an input error resulting in incomplete solution matrix (see 10 CFR 50.46 letter LTR-NRC-04-17, dated March 25, 2004). Two plant-specific calculations were found to be affected by this error. The applicant confirmed that correction of the error did not change the fundamental LOCA transient characteristics (e.g., blowdown cooling and reflood turnaround timing and behaviors). The reference double-ended guillotine break was used to develop the PCT assessments for each plant. The test simulation affected by this error was also corrected, and the transient calculation repeated. It was found that the error correction had no significant effect on the calculation results, and the prior validation conclusions remain valid. The correction is used for the AP1000 BELOCA analyses.

The second nondiscretionary change concerned Inconel 690 material properties capability (see 10 CFR 50.46 letter LTR-NRC-04-17). The material properties were revised in 2000, but the annual 10 CFR 50.46 report did not include the change. This capability was reported to correct that omission. The Inconel 690 material properties are only used for replacement steam generator analyses where the tube material has changed. The analyses directly reflect the effect of the material properties.

The staff identified the following three nondiscretionary changes with the potential to affect any of the prior code assessment and uncertainty results, and that have an effect on the AP1000 BELOCA analyses:

(1) Revised Blowdown Heatup Uncertainty Distribution

This nondiscretionary change was reported in a Westinghouse 10 CFR 50.46 letter LTR-NRC-05-20, dated April 11, 2005. This error was previously reported in a Westinghouse letter, LTR-NRC-04-11, dated February 3, 2004. As a result of input errors in the loss-of-fluid test (LOFT) facility model used to compare the predicted PCT to the test data to determine this distribution, revised analyses were performed with the version of WCOBRA/TRAC available at that time. As a result of the reanalysis with the modeling error corrections, revised blowdown heatup heat transfer multipliers were developed and the revised cumulative distribution function (CDF) was programmed into the new version of HOTSPOT. The applicant estimated the PCT effect of the revised blowdown heatup CDF by calculating the impact on the reference transient for representative two-, three-, and four-loop plants. The estimates bounded all of the 95th percentile HOTSPOT results. The applicant also made plant-specific estimates of the effect of the revised overall code uncertainty for blowdown for those plants that track the blowdown period.

The applicant identified the errors in the LOFT analyses in its response to RAI-SRP15.6.5-SRSB-03. The most important input error was the flag for the fuel rod gap pressure calculation, which was erroneously set to the steady-state option during the transient calculation. With this flag, the critical heat flux calculation was skipped and the transition to film boiling was only a result of the depletion of liquid from the core region. This was the primary input error that impacted the blowdown heatup heat transfer multiplier distribution. Other modeling aspects of the accumulator and break noding were also updated for consistency with the final, approved version of WCAP-12945-P-A, the one-dimensional pipe to three-dimensional vessel connection input was corrected, and errors in the choked flow flag applied to selected components were corrected.

The development of the blowdown heatup heat transfer multipliers and the resulting revised CDF were consistent with the previously approved methods used in WCAP-12945-P-A. The AP1000 ASTRUM analysis uses this revised CDF for the blowdown heatup heat transfer multipliers, consistent with an ASTRUM analysis of a standard Westinghouse PWR.

(2) Improved Automation of End of Blowdown Time

This discretionary change was reported in 10 CFR 50.46 letters LTR-NRC-05-20 and LTR-NRC-06-8, dated March 16, 2006. The automated selection of the end of blowdown time was first modified by replacing the criterion of 276 kilopascal (kPa) (40 pounds-force per square inch absolute (psia)) with a selection based on the time at which the system pressure stops decreasing. It was again modified by replacing the criterion related to when the system pressure stops decreasing with a selection based on the time when the collapsed liquid level in the lower plenum reaches a minimum and begins to increase again. The redefinition of the end of blowdown is a discretionary change. It has no impact on the WCOBRA/TRAC results and does not affect the prior assessments and uncertainties.

In its response to RAI-SRP15.6.5-SRSB-04, the applicant confirmed that the AP1000 BELOCA ASTRUM analysis used the improved automated method to define the end of blowdown based on the time at which the collapsed liquid level in the lower plenum reaches a minimum and begins to increase again. The time at which the collapsed liquid level is at its absolute minimum is selected as the end of blowdown time. For large double-ended guillotine breaks, similar results are obtained whether the end of blowdown is defined by RCS pressure criteria or at the time the lower plenum collapsed liquid level reaches a minimum. For the smaller breaks sampled in the ASTRUM analyses, the improved definition based on the lower plenum collapsed liquid level is more applicable than the historical pressure criterion. The ASTRUM analyses use a consistent definition of end of blowdown as the time at which the lower plenum collapsed liquid level is at its absolute minimum; this definition is applied for all runs.

(3) HOTSPOT Fuel Relocation

This nondiscretionary change was reported in 10 CFR 50.46 letter LTR-NRC-08-24, dated May 15, 2008. It was discovered that the effect of the fuel relocation on the local linear heat rate was being calculated correctly in accordance with the approved model, but then canceled out later in the coding. The HOTSPOT fuel-relocation error was an error in code logic that does not affect the approval of the fuel-relocation model. The applicant evaluated the impact of this error in letter DCP/NRC2074, "10 CFR 50.46 Report for the AP1000 Standard Plant Design," dated February 15, 2008. The impact was estimated to be ∆ 0 °Celsius (C) (∆ 0 °Fahrenheit (F)) during the blowdown phase and ∆ 38.9 °C (70 °F) during the reflood phase of the accident. The HOTSPOT fuel relocation error is a nondiscretionary change. It has no impact on the WCOBRA/TRAC results and does not affect the prior assessments and uncertainties. The AP1000 BELOCA analyses use the corrected logic.

The applicant has incorporated all of the 10 CFR 50.46 discretionary and nondiscretionary changes into the current versions of the WCOBRA/TRAC (M7AR7_AP) and HOTSPOT 6.1 codes. The discretionary changes were shown not to have a significant impact on the

WCOBRA/TRAC results and did not affect the prior assessments and uncertainties. The nondiscretionary changes addressed and corrected known errors.

15.2.6.5.2.1.2 WCOBRA/TRAC Code Validation

The phenomena identification ranking table for large-break LOCAs, provided in WCAP-15644-P, Revision 2, and NUREG-1793, indicates that the main difference between the AP1000 and operating PWRs is the DVI. Similar to validation performed for the AP600 design documented in WCAP-14171 and WCAP-14172 (nonproprietary), "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," Revision 2, additional validation was performed to examine code capability of WCOBRA/TRAC (M7AR7_AP) to model the AP1000 DVI into the downcomer. Appendix B to APP-GW-GLE-026 documents the DVI assessment calculations. Researchers validated the DVI injection by comparing the code calculations with the tests carried out on the Cylindrical Core Test Facility and the Upper Plenum Test Facility.

Calculations for the Cylindrical Core Test Facility Test 78, Run 58, showed a reasonable prediction of the thermal-hydraulic behavior with WCOBRA/TRAC (M7AR7_AP). The filling of the core and downcomer were in reasonable agreement with the data, and the calculated core level was slightly overpredicted and the downcomer level underpredicted. The maximum clad temperatures in the core were also reasonably predicted at most elevations, and they tended to be overpredicted for the higher power rods. Overall, the applicant's evaluation indicated that the WCOBRA/TRAC M7AR7_AP calculations adequately captured the thermal-hydraulic effects associated with downcomer injection, with results similar to those for the AP600.

Calculations for the Upper Plenum Test Facility Test 21, Phases A, B-I, and B-II/III, also showed that the WCOBRA/TRAC (M7AR7_AP) code reasonably and conservatively predicted the downcomer bypass phenomenon, with results similar to those for the AP600.

The applicant's assessments demonstrated that the end-of-bypass remains conservatively calculated. They also demonstrated that the PCT predictions from WCOBRA/TRAC (M7AR7_AP) remain conservative at higher fuel rod elevations and in higher power rods, and essentially best estimate at lower elevations.

Based on its review, the staff concludes that WCOBRA/TRAC (M7AR7_AP) conforms to the guidance provided in RG 1.157 and is acceptable for AP1000 BELOCA analyses to demonstrate compliance with the 10 CFR 50.46 criteria.

15.2.6.5.2.2 WCOBRA/TRAC Nodalization Model for AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Analyses

Appendix C to APP-GW-GLE-026 describes the WCOBRA/TRAC nodalization model for AP1000 BELOCA analyses. The nodalization model was developed using the methodology established in WCAP-12945-P-A. The major portion of a BELOCA analysis involves generating the plant-specific vessel and loop model for the WCOBRA/TRAC analyses. The vessel model, in particular, requires detailed information regarding the reactor pressure vessel internals. The AP1000 nodalization was also compared to the available modeling guidance presented in WCAP-16009-P-A.

15.2.6.5.2.2.1 Reactor Coolant System Loop Model

The AP1000 WCOBRA/TRAC RCS loop model (Figure C-3 in APP-GW-GLE-026) uses 75 components and 87 junctions to model the RCS loops and the passive safety systems, including the vessel module and the interface junctions between the one-dimensional loop components and the three-dimensional vessel module. The model includes the four cold legs, the two hot legs, the two steam generators, and the pressurizer.

The two passive safety injection system loops are modeled separately, complete with the CMT and balance line and the accumulator. The model also includes the in-containment refueling water storage tank (IRWST), the sump connections into the DVI lines, the PRHR, and the ADS.

The ADS modeling is simplified and excludes ADS Stages 1, 2, and 3, located on the top of the pressurizer, because ADS Stage 1 is actuated by a low-level signal from the CMTs, whose flow is effectively shut off by the actuation of accumulators. The CMT liquid level is expected to remains above the ADS Stage 1 actuation setpoint throughout the AP1000 BELOCA cladding temperature excursion, even though CMT injection begins again later in the transient. Therefore, ADS Stages 1, 2, and 3 are not expected to actuate during a large break until long after the PCT is calculated to occur. ADS Stage 4, located on the hot legs, is modeled because the limiting WCOBRA/TRAC case is extended beyond the fuel rod quench time until the CMT liquid level decreases to the low-2 setpoint that actuates the ADS Stage 4 valves and IRWST injection.

The RCS loop model also includes one-dimensional PIPE components, which model the thimble tube bypass in the low-power, support column/open hole, and guide tube assemblies. The inclusion of these components is consistent with the approved modeling of the thimble tube bypass in standard Westinghouse PWRs.

WCAP-16009-P-A describes the cold leg break nodalization used to model the break type (guillotine or split) and size.

It should be noted that Revision 17 of the DCD increased the inside diameter of the pressurizer and decreased the vessel height. However, the internal free volume of the pressurizer and the water volume, at power, remain unchanged (refer to APP-GW-GLR-016) and, therefore, have no effect on the WCOBRA/TRAC loop model. However, the decrease in the pressurizer vessel height results in the changes to the high-3, high-2, and high-1 pressurizer water level setpoints shown in DCD Table 15.0.4a. In addition, in letter DCP/NRC2074, the applicant reported that the pressurizer surge-line resistance used in the DCD Revision 15 WCOBRA/TRAC model was in error. This error has been corrected in this WCOBRA/TRAC model.

15.2.6.5.2.2.2 Reactor Pressure Vessel and Internals Model

For the AP1000 17x17 fuel, each fuel bundle contains 264 fuel rods, 24 thimble tubes, and 1 instrumentation tube. The AP1000 reactor pressure vessel model includes five fuel rod groups, based on the upper internal region design and power levels. Rod 1 represents a single fuel rod, the hot rod, which has the highest power in the core and is located in the hot assembly. Rod 2 represents the remaining 263 fuel rods in the hot assembly. It has a power equivalent to the hot assembly average fuel rods and represents an assembly located beneath a support column. Rod 3 represents the 15,576 fuel rods in the medium-power assemblies located beneath support-column/open-hole channels. Rod 4 represents the 18,216 fuel rods in the medium-power assemblies located beneath guide-tube channels. Rod 5 represents the 7,392 fuel rods in the peripheral assemblies in the low-power channel.

The location of the vertical section boundaries relative to the reactor pressure vessel structures and internals were chosen consistent with the approved AP600 WCOBRA/TRAC model described in WCAP-14171, Revision 2; NUREG-1512; and WCAP-14601, "AP600 Accident Analyses—Evaluation Models," Revision 2, with the elevations appropriate for the AP1000 design.

The model includes changes to reflect the following design changes to AP1000 reactor internals described in WCAP-16716-NP, "AP1000 Reactor Internals Design Changes," Revision 2, issued May 2007:

- relocation of radial support keys and tapered periphery on the lower core support plate (LCSP)
- addition of a flow skirt in the lower reactor vessel head
- addition of four neutron panels, attached to the outside surface of the core support barrel

The applicant has processes that identify plant configuration changes that could potentially impact safety analyses. It used these internal processes, along with internal processes for assessing evaluation model changes and errors, to identify the need to assess the impacts on LOCA analyses.

As shown in Appendix C to APP-GW-GLE-026, the downcomer region used six azimuthal sectors, as compared to four for the reference model in WCAP-16009-P-A, to account for the DVI lines, similar to the AP600 model. The two additional sectors span two of the four neutron panels, with the remaining two neutron panels falling on gaps (centered between two azimuthal sectors).

The following describes each of the three design changes in detail:

(1) Radial Support Keys/Tapered Periphery on the LCSP

The change to the AP1000 LCSP relocated the four lower radial support keys for the core barrel from the current location of 45 degrees from the cardinal axes to the cardinal locations, which eliminates the potential for interference with the core shroud attachment studs and nuts at the 45-, 135-, 225-, and 315-degree locations. The radial supports keys are now physically aligned with the locations of the cold leg nozzles.

In its response to RAI-SRP15.6.5-SRSB-06, the applicant provided a detailed description of the revised LCSP region modeling for the AP1000 WCOBRA/TRAC model. The metal volume of the LCSP was accounted for in the lower plenum channel. The radial keys were modeled in the downcomer channels to reflect the physical location of the radial keys consistent with the vessel volumes represented by the downcomer channels. One radial key was modeled in each of the two downcomer channel stacks connected to the DVI lines. One half of a radial key was modeled in each of the four downcomer channel stacks connected to the cold legs. The fraction of the radial key in each of these downcomer channels was averaged across the channel because the WCOBRA/TRAC code was not designed to reflect more detailed azimuthal modeling than that specified by the user through the nodalization and lateral gap connections. The momentum area and loss coefficients were adjusted to calibrate the steady-state pressure drop consistent with the modeling approach used for standard plants and the resolution of the WCOBRA/TRAC model nodalization. The AP1000 ASTRUM steady-state calculation confirmed that the steady-state acceptance criteria, specified in WCAP-16009-P-A, Table 12-6, "Criteria for Acceptable Steady-State," were met. The vessel pressure drop and the vessel inlet nozzle to mid-core pressure drop were benchmarked against hydraulic calculations to be within the acceptance criteria.

(2) Lower Reactor Vessel Head Flow Skirt

One of the design changes to the AP1000 reactor internals involved the addition of a flow skirt in the lower reactor vessel head to obtain a more uniform core inlet flow distribution that meets specifications established by the Westinghouse fuel group. The applicant provided a detailed description of the flow skirt region modeling for the AP1000 WCOBRA/TRAC model in response to RAI-SRP15.6.5-SRSB-07. The metal volume of the flow skirt was accounted for in the lower plenum channel. The radial flow area through the flow skirt holes and between the top of the flow skirt and the bottom of the LCSP was reflected in the gaps that connect the downcomer channels to the lower plenum channel. The AP1000 ASTRUM steady-state calculation confirmed that the steady-state acceptance criteria, specified in WCAP-16009-P-A, Table 12-6, were met. The vessel pressure drop and the vessel inlet nozzle to mid-core pressure drop were benchmarked against hydraulic calculations to be within the acceptance criteria. The applicant determined that the core inlet flow distribution assumptions that formed the basis of the original fuel thermal-hydraulic calculations remain valid.

(3) Neutron Panels

Neutron panels were attached to the outside diameter of the core support barrel to maintain the end-of-life reactor vessel fluence values at less than the maximum allowed in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, issued May 1988. The panels reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values and provide a relatively rigid structure with a smaller downcomer cross-sectional area than a full cylinder. The neutron panels are located at four circumferential locations where fuel assemblies are closest to the reactor vessel (the 0-, 90-, 180-, and 270-degree locations). The DVI lines are located at 0 and 180 degrees, and the hot leg nozzles are located at 90 and 270 degrees. Each neutron panel covers about 30 degrees circumferentially and extends over the entire length of the active core region (4.27 m (14 ft)). The neutron panels are contoured to minimize the impact on the downcomer annulus flow area reduction and to reduce the probability of vortex generation in the downcomer.

The applicant provided a detailed description of the neutron panel modeling for the AP1000 WCOBRA/TRAC model in response to RAI-SRP15.6.5-SRSB-09. The neutron panels were modeled in the downcomer channels to reflect the physical location of the neutron panels consistent with the vessel volumes represented by the downcomer channels. One neutron panel was modeled in each of the two downcomer channel stacks connected to the DVI lines. One half of a neutron panel was modeled in each of the four downcomer channel stacks connected to the cold legs. The fraction of the neutron panel in each of these downcomer channels was averaged across the channel because the WCOBRA/TRAC code was not designed to reflect more detailed azimuthal

> modeling than that specified by the user through the nodalization and lateral gap connections. The azimuthal flow areas were modeled as the flow area away from the neutron panels, and the friction factor for azimuthal flow reflects the flow between two walls, consistent with the modeling approach for standard plants. The metal mass of the neutron panels was modeled as an unheated conductor in the appropriate channels. This is an acceptable approach for capturing the effects of the neutron panels in the downcomer, including the calculation of downcomer boiling, and is consistent with the modeling of neutron panels in standard two-loop, three-loop, or four-loop Westinghouse PWRs.

15.2.6.5.2.2.3 Previous WCOBRA/TRAC Modeling Limitation

The AP600 safety evaluation in NUREG-1512, on the acceptability of the AP600 WCOBRA/TRAC model, imposed limitations that require an applicant to address the sensitivity of the CMT and residual heat removal system modeling parameters that are not included in the uncertainty methodology in the event that either the blowdown or reflood phase PCT exceeds 941 °C (1725 °F) for any reason. The applicant addresses this by repeating the study that identifies the PCT sensitivity to CMT/PRHR elimination and adding the blowdown and reflood PCT impacts as a bias to their respective 95 percent PCT results. Section 15.2.6.5.2 of NUREG-1793 also discussed this limitation. Revision 17 of the DCD, Section 15.6.5.4A.5, states that previous AP1000 sensitivity calculations evaluated the sensitivity of the CMT and PRHR to modeling relative to a baseline case. The results show that the calculated PCTs for both the cases, in which the CMT and the PRHR, respectively, were isolated from the rest of the AP1000, were lower than the PCT of the baseline case. In RAI-SRP15.6.5-SRSB-12, the NRC asked the applicant to clarify whether the sensitivity studies for the proposed AP1000 model include the MLO and CWO sensitivities. In its response, the applicant performed sensitivity studies for the CMT and PRHR with the model described in Appendix C to APP-GW-GLE-026. The WCOBRA/TRAC PCT results of the CMT inoperable study showed a temperature decrease of Δ 21 °C (38 °F) when compared to the reference case. The results of the PRHR inoperable study showed a temperature increase of Δ 1.1 °C (Δ 2 °F) when compared to the reference case. However, to perform the PRHR study, the maximum allowable time step had to be reduced by 0.0001 second to execute the case, and a revised reference case was also run with the reduced time step. The revised reference case resulted in a decreased PCT of Δ 3.3 °C (Δ 6 °F) as compared to the original reference case. The applicant concluded that the effect of PRHR inoperability was minimal. Since the AP1000 shows significant margin to the MLO and CWO limits, the staff concludes that no penalties need to be applied to the analysis results for assuming that the safety-related equipment does not operate.

In Attachment 2 to APP-GW-GLE-026, the applicant described the following revisions to Revision 17 of the DCD, Section 15.6.5.4A.5, “Large-Break LOCA Analysis Results,” to incorporate the results of these studies:

> The large break LOCA analysis complies with the restrictions in Reference 32 [WCAP-16009-P-A]. AP1000 sensitivity calculations evaluated the sensitivity to modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 1 ºC (2 °F) higher than the reference transient configuration.

This is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In summary, the AP1000 ASTRUM BELOCA model was developed consistent with the modeling guidelines presented in WCAP-16009-P-A. Therefore, the staff concludes that the AP1000 ASTRUM BELOCA model using WCOBRA/TRAC (M7AR7_AP) conforms to the guidance provided in RG 1.157 for evaluation models needed to demonstrate compliance with the acceptance criteria in 10 CFR 50.46.

15.2.6.5.2.3 ASTRUM AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Analysis

15.2.6.5.2.3.1 ASTRUM Applicability to AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation

ASTRUM uses a nonparametric statistical technique applied directly to a random sample of outputs, for example, the PCT, the MLO, and the CWO. These sample outputs are computed by applying Monte Carlo sampling of the inputs to the WCOBRA/TRAC calculations. With this approach, a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of 10 CFR 50.46 can be developed. Once the desired tolerance level is defined, the number of Monte Carlo code runs required to construct the tolerance interval that meets the desired level of safety can be computed. ASTRUM is based on a 95/95 tolerance level to demonstrate compliance with the 10 CFR 50.46 criteria. This tolerance level requires 124 WCOBRA/TRAC runs.

A confidence interval covers a population parameter with a stated confidence, that is, a certain proportion of the time. A fixed proportion of the population can also be covered with a stated confidence, called a tolerance interval. The endpoints of a tolerance interval are called tolerance limits. An application of tolerance intervals involves comparing specification limits with tolerance limits that cover a specified proportion of the population. The 95/95 statement is interpreted to mean that there is a 95 percent probability that 95 percent of the random output variable population falls within the specified tolerance limits. The maximum and minimum value of the samples of the output variables are used to define the limits. The results obtained with ASTRUM are taken to mean that there is at least a 95 percent confidence that the limiting PCT, MLO, and CWO from the sample exceed the true 95th percentile.

ASTRUM replaced the original CQD response surface method, described in WCAP-12945-P-A, to combine the uncertainties with a direct Monte Carlo sampling method. In the application of CQD methodology to AP1000, the uncertainties in the initial fluid conditions in the RCS and the ECCS boundary conditions were bounded in a direction to maximize the PCT. The model uncertainty component addressed uncertainties in the code models that affected the overall system transient (global models), as well as those that only affected the hot rod (local models). WCOBRA/TRAC was used to calculate the effects of initial conditions, power distributions, and global models, and HOTSPOT was used to calculate the effects of local models. Biases and uncertainties, resulting from the assumption that the initial conditions, the power distribution, and the model uncertainty components were linearly combined, were quantified and taken into account. The CQD methodology calculates the final PCT uncertainty distribution by a combination of response surface equations and Monte Carlo sampling. ASTRUM considers the same plant parameters, but each parameter is randomly sampled for each case. The 95/95 PCT is established using nonparametric order statistics.

With ASTRUM, the number of runs is fixed (124 runs for three outcomes—PCT, MLO, and CWO) and is independent of the number of uncertainty attributes considered in the sampling process. The uncertainty parameters are directly sampled instead of using the bounding approach of the CQD methodology.

ASTRUM retains the distinction between global and local variables. However, in ASTRUM, only a single HOTSPOT calculation is performed for each WCOBRA/TRAC run, instead of the multiple HOTSPOT runs with the CQD methodology used to obtain the local model PCT distribution. The HOTSPOT calculation is now a single calculation where the local uncertainties are set at their values by random sampling from their respective distributions. This is consistent with the Monte Carlo approach, where each uncertainty parameter is randomly sampled from the respective distribution for each simulation, which comprises a WCOBRA/TRAC and a HOTSPOT calculation.

The NRC reviewed the ASTRUM uncertainty methodology and found it to be acceptable for meeting the regulatory requirements of 10 CFR 50.46, as described in the staff evaluation of WCAP-16009-P-A. The AP1000 BELOCA analyses for Revision 17 of the DCD use the previously approved global model uncertainties and biases, as well as the local model uncertainties and biases, including the revised blowdown heatup transfer multiplier. The ASTRUM uncertainty methodology is independent of the physical system being modeled and is equally applicable to the AP1000 BELOCA analyses to demonstrate compliance with the requirements of 10 CFR 50.46.

15.2.6.5.2.3.2 Application of ASTRUM to the AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation

The ASTRUM uncertainty methodology, used for the AP1000 BELOCA analysis for Revision 17 of the DCD, independently samples the uncertainties of the global models, local models, power distribution, and initial and boundary conditions for each of 124 runs over the same ranges of uncertainty and distributions as in the CQD methodology. The sampled uncertainties become inputs to each of the 124 WCOBRA/TRAC calculations.

The WCOBRA/TRAC thermal-hydraulic boundary conditions for the hot rod are input to the local model uncertainties calculation performed with HOTSPOT. The limiting PCT, MLO, and CWO may come from the same case or as many as three different cases. With ASTRUM, each parameter is assumed to be independent of the other two parameters. This assumption is conservative since the MLO and the CWO depend on the cladding temperature (time at temperature).

The WCOBRA/TRAC studies were performed for the AP1000 BELOCA to determine the sensitivities to some of the major plant parameters. These studies included effects of ranging the steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and the break location. The results were used to identify bounding conditions for these parameters, which were then used in the uncertainty calculations.

APP-GW-GLE-026, Table E-1, "Summary of Plant Physical Description, Initial Conditions, Power Distribution, and Global Model Uncertainty Application in CQD Methodology, AP1000 DCD Methodology, and ASTRUM Methodology as Applied to the AP1000," compares the plant physical models, initial conditions, power distributions, and global model uncertainties between the CQD method used in Revision 15 of the DCD and ASTRUM used in Revision 17. The values were referenced to the appropriate information in WCAP-16009-P-A.

DCD Table 15.0-4a provided the PMS setpoints and time delay assumed in the accident analyses. As described in Section 15.2.6.5.2.2 of this report, the safety system related to pressurizer water level high-3, high-2, and high-1 setpoints in DCD Table 15.0-4a are consistent with the revised pressurizer dimensions described in APP-GW-GLR-016.

Revision 17 of the DCD, Table 15.6.5-4, provides the major plant parameter assumptions used in the BELOCA analysis. These plant parameters were developed consistent with the guidance provided in WCAP-16009-P-A. The uncertainty distributions for the global models, local models, power-related parameters, and initial and boundary conditions appear in Tables 1, 2, 3, and 4, respectively, of APP-GW-GLE-026. These distributions are consistent with Tables 1-7, 1-8, 1-10, and 1-11, respectively, of WCAP-16009-P-A. The blowdown heatup heat transfer multiplier was modified to correct modeling inconsistencies and input errors in the WCOBRA/TRAC LOFT deck, as described in Section 15.2.6.5.2.1 of this report.

The initial reactor core power of less than 1.01x3400 MWt assumed a calorimetric uncertainty of 1 percent. This is consistent with DCD Table 15.0-2, which lists the initial thermal power output assumed for the large-break LOCA analysis as 3,434 MWt. As stated in the staff safety evaluation in Section 15.1.0.3.1.1 of this report, the applicant will revise DCD Section 15.0.15 to include COL Information Item 15.0.15.1, which requires the COL holder to calculate the primary power calorimetric uncertainty before fuel load to confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated value. Therefore, the staff finds the use of 1 percent initial core power uncertainty acceptable.

The ranges of the parameters were compared to the TS limiting conditions for operation. The accumulator water temperature is ranged based on AP1000 expected values from 10 °C (50 °F) to 48.9 °C (120 °F), which is consistent with the TS 3.6.5, “Containment Air Temperature,” limiting condition for operation for an operable containment.

The accumulator pressure range between 4619 kPa (670.0 psia) and 5280 kPa (765.8 psia) was inconsistent with TS 3.5.1, SR 3.5.1.3, which specifies the accumulator pressure range between 4392 kPa and 5302 kPa (637 and 769 pounds-force per square inch gauge (psig)). In addition, Table 15.6.5-4 did not include the accumulator water volume range. In its response to RAI-SRP15.6.5-SRSB-13, the applicant indicated that the incorrect accumulator pressure and water volume ranges were assumed in the AP1000 BELOCA ASTRUM analysis, and a reanalysis of the top 10 HOTSPOT PCT cases from the ASTRUM run set was performed to evaluate the impact of the incorrect pressure and water volume ranges. Maintaining the seed values used in the original ASTRUM analyses, the accumulator pressures and liquid volumes for the top 10 HOTSPOT cases were determined based on the original sampling for each run and the revised ranges. The evaluation showed that the PCT, MLO, and CWO results reported in APP-GW-GLE-026 remained valid for the revised TS ranges. The applicant assessed a 0-degree PCT penalty for this set of closely related errors. It must be noted that Figure 27, “HOTSPOT PCT Versus Effective Break Area Scatter Plot for All 124 Cases,” in APP-GW-GLE-026 would likely show some changes in the PCT values for some cases if the complete set of 124 runs were to be reanalyzed with the correct accumulator pressure and water volume ranges. However, the overall conclusions for the limiting break would still be applicable.

The applicant also committed to update Table 15.6.5-4 of the DCD to make the accumulator pressure range and the accumulator water volume ranges consistent with TS 3.5.1, SR 3.5.1.3 and SR 3.5.1.2, respectively. In a subsequent revision to the DCD, the applicant incorporated this change in the DCD.

The percentage of steam generator tube plugging is based on the bounding case (10 percent), and DCD Table 5.1-3, “Thermal-Hydraulic Parameters (Nominal),” provides the nominal RCS thermal-hydraulic parameters for this case. The average RCS temperature (T_{AVE}) listed in Table 15.6.5-4 is consistent with DCD Table 5.1-3 for this level of plugging. The hot assembly location, under a support column, is bounding based on the design of the reactor vessel upper core region internals. The pressurizer location, in the intact loop, is bounding based on AP1000 WCOBRA/TRAC sensitivity calculations.

The single-failure assumption is the failure of one CMT isolation valve to open. During a large-break LOCA, the CMT is actuated by the ”S” signal, but the CMT flow is shut off upon accumulator actuation and resumes at the end of accumulator injection. This limiting single failure affects the safety injection flow delivery when the CMT begins to inject toward the end of accumulator injection.

One of the plant parameter assumptions is that the offsite power remains available during the LOCA transient. In its response to RAI-SRP15.6.5-SRSB-14, the applicant provided an offsite power availability/unavailability sensitivity analysis using the AP1000 BELOCA model. It confirmed that the offsite-power-available assumption remains limiting in the AP1000 ASTRUM analysis, because of the effect of RCP operation on the downflow cooling during the blowdown period. With loss of offsite power, the RCPs trip coincident with the break and begin to coast down. With offsite power available, the RCPs continue to run until they are automatically tripped. As a result, more fluid flows from the upper plenum into the hot leg in the broken loop, and less fluid reverses flow from the hot leg of the intact loop into the upper plenum. Therefore, less fluid in the upper plenum is available for blowdown cooling. In the case of a loss of offsite power, the liquid downflow into the hot assembly was observed during blowdown; however, there was no observed downflow of liquid into the hot assembly at the upper core plate with offsite power available. The downflow cooling in the case of a loss of offsite power results in increased blowdown cooling and lower PCT.

The containment backpressure is specified at a bounding minimum value, consistent with the WCAP-16009-P-A methodology. The containment pressure is specified at the break location as an input table. The AP1000 ASTRUM analysis followed the approved WCAP-16009-P-A methodology for determining the conservative containment backpressure approved for standard Westinghouse PWRs. The reference transient was used to establish the containment pressure response that was applied as a boundary condition in the uncertainty analysis calculations. The inputs to the containment pressure calculation were biased to obtain a conservative (low) pressure transient.

The peak linear heat rate, expressed in terms of total heat flux hot channel factor, F_Q, was compared to AP1000 DCD Table 4.3-2, “Nuclear Design Parameters (First Cycle),” and was found to be consistent with the design data. The hot rod assembly power, expressed in terms of nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, was increased from 1.65 to 1.75. In its response to RAI-SRP15.6.5-SRSB-17, the applicant clarified that it increased $F_{\Delta H}$ in the AP1000 ASTRUM analysis in order to provide increased margin for the core design. The hot assembly average integrated power (P_{HA}) was correspondingly increased to 1.683 from 1.586. The hot assembly average integrated power is 4 percent lower than the hot rod integrated power, consistent with the standard value applied in the approved ASTRUM, WCAP-16009-P-A. DCD Figure 15.6.4A-13 shows the axial power distribution in terms of normalized power integrals in the bottom third of the core and middle third of the core, which is consistent with the treatment in WCAP-16009-P-A.

Only cold leg breaks were analyzed because the hot leg break location was found to be nonlimiting for the BELOCA methodology. ASTRUM explicitly accounts for the effect of break type, and various break types and sizes, such as double-ended cold leg guillotine (DECLG) and split breaks, are assumed to have an equal chance of being sampled. The break size and type were sampled consistent with this methodology. The applicant addressed the limiting break type analysis in its response to RAI-SRP15.6.5-SRSB-14. Before performing the detailed ASTRUM uncertainty analyses, confirmatory calculations were performed to identify the limiting settings for some of the major plant parameters. The results of these calculations were used to define the reference transient case. The applicant found that the important thermal-hydraulic characteristics of the AP1000 during a large-break LOCA were consistent with those observed in conventional Westinghouse three-loop plant analyses. The AP600 large-break LOCA phenomena identification ranking table indicated that the only high-ranking area of difference between the AP600 and a standard three-loop plant with respect to a large-break LOCA was the delivery of emergency core cooling water through the AP600 DVI lines (see the WCAP-14171, Revision 2, and WCAP-15644, Revision 2, WCOBRA/TRAC code applicability reports for the AP600 and the AP1000, respectively). The applicant's experience with BELOCA analyses has shown that, for three-loop plants, either a split break or a double-ended guillotine break may be limiting. The reference transient configuration was determined from simulations of nominal DECLG breaks consistent with the approved ASTRUM for conventional three-loop plants (WCAP-16009-P-A).

For Revision 15 of the DCD, the limiting large-break LOCA analyzed with the CQD methodology was determined to be a DECLG break. In Revision 17, the combination of uncertainty parameters sampled in the ASTRUM analysis resulted in the limiting break being a split break, which is the limiting case for both PCT and MLO. Figure 27 of APP-GW-GLE-026 depicts this result and provides the PCT scatter plot showing the impact of the effective break area on the HOTSPOT PCT analysis.

DCD Section 15.6.5.4A.6 describes the limiting PCT/MLO split break case from the AP1000 ASTRUM analysis with the results shown in Figures 15.6.5.4A-1 through 15.6.5.4A-12. Figure 15.6.5.4A-2 shows that the HOTSPOT PCT occurs during the reflood phase. In its response to RAI-SRP15.6.5-SRSB-14, The applicant confirmed that the AP1000 remains reflood limited and that the design changes did not result in the limiting PCT moving to the blowdown time period. The applicant inspected the 124 ASTRUM calculations. In 100 runs, the PCT occurred during reflood. The applicant then inspected the characteristics of the 24 runs in which the PCT occurred during blowdown. These 24 runs were significantly nonlimiting, with the WCOBRA/TRAC PCTs less than 649 °C (1200 °F). The effective break areas sampled for 21 of these calculations were less than 1.0 times the cold leg area, and one was a split break with an effective break area greater than 2.0. The peaking factor and power shape parameters sampled in the remaining two cases contributed to the low PCTs in these calculations. Overall, the early PCT peak in these 24 runs was attributed to the run-specific combinations of sampled parameters, particularly the sampled break areas for the majority of the runs. The applicant determined that the ASTRUM runs in which the calculated PCT peaks occurred early in the transient resulted from the combination of sampled parameters for those runs and not the AP1000 design changes.

DCD Table 15.6.5-6, "Best-Estimate Large-Break Sequence of Events for the Limiting PCT/MLO Case," provides the time line for the sequence of events for the limiting BELOCA. The containment high-2 pressure (156 kPa (22.7 psia or 8 psig)) is assumed to occur 2.2 seconds after the initiation of the break because the massive size of the break causes an immediate, rapid pressurization of the containment. This assumption of 2.2 seconds is consistent with the time delay for the "S" signal on high-2 containment pressure for a large-break LOCA specified in DCD Table 15.0-4a. In its response to RAI-SRP15.6.5-SRSB-15, the applicant addressed the acceptability of the assumed containment response to the BELOCA spectrum. The 2.2-second assumption overestimates the time to reach the containment pressure high-2 setpoint for a nominal DECLG large-break LOCA. The applicant's calculations with assumptions biased to obtain a conservatively low pressure transient show the containment pressure to be more than 165 kPa (24 psia), as compared to the 156 kPa (22.7 psia or 8 psig)) high-2 setpoint, at 2.2 seconds after the break. While 2.2 seconds may not allow enough time to reach the high-2 containment pressure setpoint for the smallest breaks sampled as part of ASTRUM because of the reduced mass and energy release, these smaller break sizes were found to be nonlimiting for the AP1000 as shown in APP-GW-GLE-026, Figure 27.

DCD Section 15.6.5.4A.5 states that local and core-wide cladding oxidation values have been determined using the methodology approved in WCAP-16009-P-A. The detailed CWO calculation procedure described in Section 11.6-2 of WCAP-16009-P-A would require a series of <u>W</u>COBRA/TRAC runs based on the oxidation value from the limiting hot assembly rod case, but with varying power levels for the other runs in the series to account for lower power assemblies in other core regions. For the AP1000, the applicant chose to apply the limiting case hot assembly rod oxidation value to the entire reactor core. In its response to RAI-SRP15.6.5-SRSB-18, the applicant clarified that the method used to evaluate the limiting CWO is a conservative approach and in accordance with the procedure used for standard ASTRUM analyses. The results of the 124 cases were ranked by the <u>W</u>COBRA/TRAC-calculated hot assembly average volumetric oxidation from highest to lowest. The case with the maximum hot assembly average volumetric oxidation was examined. In the AP1000 ASTRUM analysis, this was case "run015" with a hot assembly oxidation of 0.2 percent. Since this oxidation was substantially lower than the regulatory CWO limit of 1 percent, a detailed calculation of the CWO was not necessary for the AP1000. By definition, the CWO fraction is less than the hot assembly average volumetric oxidation because the many lower power assemblies present in the reactor core will have lower average volumetric oxidation than the hot assembly. The application of the limiting hot assembly rod oxidation value to the entire core is conservative and acceptable.

15.2.6.5.2.4 Summary of the Best Estimate Large Break Loss-of-Coolant Accident Analysis Results

The results of the AP1000 BELOCA ASTRUM analyses, based on the 124 <u>W</u>COBRA/TRAC and HOTSPOT runs, identify the limiting PCT/MLO split break. DCD Table 15.6.5-8 summarizes the AP1000 ASTRUM BELOCA results and shows the calculated 95th percentile PCT to be 1003 °C (1837 °F), the MLO to be 2.25 percent, and the limiting CWO to be 0.2 percent. Table 15.6.5-8 also indicates that the core remains coolable and the core remains cool in the long term. The core coolable geometry is generally satisfied when calculated PCT and MLO are within the associated acceptance criteria. The post-LOCA long-term core cooling, as described in DCD Section 15.6.5.4C, "Post-LOCA Long-Term Cooling," is not affected by the application of ASTRUM and continues to show compliance with 10 CFR 50.46 acceptance criteria. Therefore, the BELOCA analysis results show a high level of probability that the following criteria of 10 CFR 50.46 will be met:

- The calculated PCT will not exceed 1204 °C (2,200 °F).
- The calculated maximum cladding oxidation will not exceed 17 percent of the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam will not exceed 1 percent of the amount that would be generated if the entire cladding metal surrounding the fuel (excluding the cladding surrounding the plenum volume) were oxidized.
- The calculated changes in core geometry are such that the core remains amenable to cooling.
- After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value, and the decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The calculated results for the AP1000 BELOCA meet the acceptance criteria of 10 CFR 50.46 and are, therefore, acceptable.

15.2.6.5.2.5 Conclusion

Based on the above evaluation, the staff concludes that the BELOCA analysis using the WCOBRA/TRAC code and the ASTRUM statistical uncertainty treatment methodology described in Revision 17 of the DCD demonstrates compliance with the acceptance criteria of 10 CFR 50.46 and, therefore, is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made appropriate changes to the DCD text, which resolves this issue.

15.2.7 Post-Loss-of-Coolant Accident Long-Term Cooling (DCD Tier 2, Section 15.6.5.4C)

DCD Section 15.6.5.4C did not change; however, the NRC is reviewing the long-term cooling analyses and supporting documentation submitted for resolution of COL Item 6.3.8.2 related to Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR Sumps,” for the AP1000, together with the GSI-191 issues addressed in Section 6.2.1.8, “Adequacy of IRWST and Containment Recirculation Screen Performance.” For consistency, the staff’s evaluation and conclusion regarding post-LOCA long-term cooling for the AP1000 appear with the GSI-191 discussion in Section 6.2.1.8 of this report.

15.3 Radiological Consequences of Accidents

In DCD Tier 2, Chapter 15, the applicant performed radiological consequence assessments of the following seven design-basis accidents (DBAs), using the hypothetical set of atmospheric dispersion factors (χ/Q values) provided in DCD Tier 1, Table 5.0-1, “Site Parameters”; DCD Tier 2, Table 2-1, “Site Parameters”; DCD Tier 2, Table 15A-5, “Offsite Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis”; and Table 15A-6, “Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis:

(1) LOCA
(2) fuel-handling accident (FHA)
(3) main steamline break accident outside of containment
(4) RCP shaft seizure accident
(5) failure of small lines carrying primary coolant outside containment
(6) rod cluster assembly ejection accident, and
(7) steam generator tube rupture accident

The applicant concluded in DCD Tier 2, Revision 15, that the AP1000 design will provide reasonable assurance that the radiological consequences resulting from any of the above DBAs will fall within the offsite dose criterion specified in 10 CFR 50.34(a)(1) and the control room operator dose criterion specified in GDC 19, “Control Room,” of Appendix A to 10 CFR Part 50. In NUREG-1793, the staff performed independent radiological consequence analyses for all of the DBAs listed above and verified the applicant’s assessments.

In September 2008, the applicant submitted Revision 17 of the DCD, as a part of its application to amend the AP1000 DC rule in Appendix D to 10 CFR Part 52. In Revision 17, the applicant requested the following four standard changes to the Chapter 15 of the certified AP1000 DCD, Revision 15:

(1) increase in the assumed decay time in the FHA dose analysis from 24 hours to 48 hours to provide increased radioactive decay of short-lived fission products before the handling of irradiated fuel assemblies

(2) increase in the aerosol removal duration in the containment from 15.5 hours to 24 hours from the initiation of a DBA

(3) revisions to some of the hypothetical offsite and control room χ/Q values, and

(4) changes to the MCR emergency habitability system (VES) operation

The applicant proposed these changes in accordance with the change criterion in 10 CFR 52.63(a)(1)(vii) in that the changes contribute to increased standardization of the certification information.

Subsequent to the submittal of the AP1000 DCD Revision 17, on November 3, 2008, the applicant proposed to add a first-of-a-kind passive control room air filtration line to the MCR VES. This design change is intended to allow the VES to meet the dose acceptance criterion specified in GDC 19 with an allowable control room unfiltered air inleakage of 25.5 cubic meters per hour (m^3/hr) (15 cubic feet per minute (ft^3/min)), which includes 8.5 m^3/hr (5 ft^3/min) for ingress/egress. Section 6.4 of this report provides the staff’s evaluation of the MCR VES, including the proposed design changes.

These changes will alter the calculated radiological doses in the control room for the above DBAs analyzed in the certified AP1000 DCD, Revision 15, and will revise some of the χ/Q values listed as site parameters in the following tables:

- DCD Tier 1, Table 5.0-1, “Site Parameters”

- DCD Tier 2, Table 2-1, “Site Parameters”

- DCD Tier 2, Table 15A-5, "Offsite Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"
- DCD Tier 2, Table 15A-6, "Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"

Revision 17 of the DCD, Section 2.3.4, evaluates the proposed changes to the hypothetical short-term accident χ/Qs, and Table 2.3.4-1 lists the accident χ/Q values used as site parameters.

15.3.1 Evaluation

15.3.1.1 Fuel-Handling Accident Decay Time Increase

In Revision 15 of the DCD, Section 15.7.4, "Fuel Handling Accident," the applicant analyzed the radiological consequences of a postulated FHA inside containment and in the fuel-handling area inside the auxiliary building, assuming that a single fuel assembly that has undergone 24 hours of decay time is dropped. This decay time provides for radioactive decay of short-lived fission products to reduce the radiological consequences of a release for the design-basis FHA. Even though the applicant analyzed the FHA using a decay time of 24 hours in Revision 15 of the DCD, the applicant had conservatively specified a 100-hour decay time in Revision 15 of the DCD, Chapter 16, TS 3.9.7, "Decay Time." The longer decay time would result in further reduction of radiological consequences of the FHA than the applicant calculated in the FHA radiological consequences analysis in Revision 15 of the DCD.

As discussed in NUREG-1793, the staff reviewed the applicant's analysis and performed an independent confirmatory radiological analysis for the postulated FHA. The staff's results agree with the applicant's values. Both the applicant's and the staff's results met the relevant dose acceptance criteria at the exclusion area boundary (EAB), low-population zone (LPZ), and control room. Based on its review, the staff finds the applicant's analysis of the FHA to be acceptable and that the dose criteria in 10 CFR 50.34(a)(1) and GDC 19 are met.

In Revision 17 of the DCD, the applicant proposed to change the minimum decay time to 48 hours from the 24 hours assumed in Revision 15 of the DCD and in NUREG-1793. This change will provide additional radioactive decay of short-lived fission products, which would result in a lesser amount of radioactive material available for release and, therefore, lower dose results for the radiological consequence assessment for the postulated FHA. Revision 17 of the DCD, Section 15.7.4, discussed the revised FHA analysis assuming a 48-hour decay time. The staff verified that the only changes to the analyses were related to the change in decay time and the changes in the hypothetical χ/Q values discussed below in Section 15.3.1.3.

The staff finds that the change in the FHA radiological consequence analysis ensures that the decay time assumption in the analysis is consistent with the minimum decay time value proposed for the AP1000 TS and results in lower control room and offsite doses than previously found acceptable for Revision 15 of the DCD. Therefore, the staff finds that the proposed change to the TS decay time is acceptable with respect to the radiological consequences of DBAs. The resulting doses off site and in the control room continue to meet the dose acceptance criteria in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000, and NUREG-0800

Section 15.0.3 for the DC review. The applicant's revised FHA dose analyses show that the dose criteria in 10 CFR 50.34(a)(1), 10 CFR 52.47(a)(2)(iv), and GDC 19 would be met with the change in minimum decay time before fuel movement. The applicant revised the decay time requirement in Revision 17 of the DCD, TS 3.9.7, to 48 hours from 100 hours to be consistent with the revised FHA analysis. The staff finds that the proposed change to the TS is supported by the FHA analysis and, therefore, is acceptable.

15.3.1.2 Aerosol Removal Duration in Containment

In Table 15.6.5-3,"Radiological Consequences of a Loss-of-Coolant Accident with Core Melt," of Revisions 15 and 17 of the DCD, the applicant provided the LPZ doses as 0.238 sievert (Sv) and 0.234 Sv (23.8 and 23.4 roentgen equivalent man (rem)) total effective dose equivalent (TEDE), respectively. In RAI-SRP15.6.5-RSAC-01, the staff asked the applicant to explain the discrepancy in these two LPZ doses, since none of the analysis assumptions given in the certified Revision 15 appear to have changed in Revision 17. In its response, the applicant stated that the analyses in Revision 17 of the DCD took full credit for aerosol removal for the first 24 hours (rather than for 15.5 hours as in Revision 15), using the removal coefficients presented in Appendix 15B to Revision 15 of the DCD. This resulted in a small decrease in the LPZ dose to 0.234 Sv (23.4 rem) TEDE. The staff had previously found the aerosol removal coefficients credited for the first 24 hours in Appendix 15B to Revision 15 of the DCD to be acceptable. Revision 17 did not change any of the aerosol removal coefficients in Appendix 15B. The staff finds that the response to RAI-SRP15.6.5-RSAC-01 is acceptable; therefore, this RAI is closed and the changes to the assumed duration of the aerosol removal in containment are acceptable.

15.3.1.3 Offsite and Control Room χ/Q Values

15.3.1.3.1 Control Room χ/Q Values

In Revision 16 of the DCD, the applicant increased the control room χ/Q values from those specified in Revision 15. In Revision 17, the applicant proposed a change in the control room isolation logic so that the increased χ/Q values in Revision 16 could be maintained, with two exceptions.

The χ/Q values for the heating, ventilation, and air conditioning (HVAC) intake at the ground level containment release points in the 2–8-hour and 8–24-hour intervals are reduced for the LOCA with the nuclear island nonradioactive ventilation system in supplemental filtration mode.

In addition, in Revision 17 of the DCD, the applicant: (1) revised the control room χ/Q values for plant vent/passive containment system air diffuser and ground level containment releases to the control room HVAC intake and Annex Building door; and (2) added new control room χ/Q values for condenser air removal stack releases to the HVAC intake and Annex Building door. Revision 17 of the DCD, Table 2.3.4-1, lists these revisions. Section 2.3.4 of this report provides the staff's evaluation and acceptance of these revisions.

15.3.1.3.2 Offsite χ/Q Values

Revision 17 of the DCD does not change the offsite χ/Q values provided in Revision 15.

15.3.1.4 Offsite and Control Room Doses

15.3.1.4.1 Offsite Doses

The applicant has not changed the radiological doses calculated for the postulated LOCA from those in Revision 15 of the DCD, except for the small decrease for the LPZ dose evaluated in Section 15.3.1.2 of this report. For the radiological consequences of DBAs other than the LOCA, the applicant used the higher and more restrictive χ/Q values provided in Revision 17 of the DCD (compared to those χ/Q values used in Revision 15), while still meeting the dose acceptance criteria. The staff verified and confirmed the applicant's revised offsite doses for the DBAs other than the LOCA. The radiological consequence analyses for the five remaining DBAs (other than the LOCA and FHA as discussed above) have not changed in Revision 17 of the DCD from those described in Revision 15.

Therefore, the DBA analyses in Revision 17 of the DCD used two sets of χ/Q values, one for the LOCA and the other for the DBAs other than the LOCA, although Table 15.A-5 shows only one set of χ/Q values used or LOCA. In its response to RAI-SRP15.6.5-RSAC-02, the applicant stated that Table 15A-5 of a future DCD revision will include these two sets of χ/Q values.

Subsequently, the applicant indicated during the November 3, 2008, meeting with the staff that it will use the same offsite χ/Q values for all DBAs in future DCD revisions to avoid having two sets of χ/Q values, one for the LOCA and another for the DBAs other than the LOCA. In a subsequent revision to the AP1000 DCD, the applicant revised the offsite doses for all DBAs except the LOCA, which resolves this issue.

15.3.1.4.2 Control Room Doses

In Revision 17 of the DCD, the applicant also proposed to modify the control room isolation to include switching to the main control room VES on the pressurizer low-pressure signal. In the LOCA dose analysis, this modification results in actuation of the VES by the time core activity releases start at 10 minutes with leak-before-break approval. In addition, the applicant proposed an effective unfiltered air inleakage assumption of 2.55 m^3/hr (1.5 ft^3/min) into the control room based on a total inleakage of 8.5 m^3/hr (5 ft^3/min), with credit taken for purging of the vestibule door volume and the incomplete mixing of the vestibule and control room volumes with outside air following ingress/egress.

During the November 3, 2008, meeting with the staff, the applicant presented the addition of a new passive control room air filtration line to the VES. This design change is intended to allow the VES to meet the dose acceptance criterion specified in GDC 19 with an allowable unfiltered air inleakage of 15 ft^3/min instead of the assumption of 5 ft^3/min total unfiltered inleakage as discussed above.

To evaluate the revised control room passive filtration design, the staff performed an independent radiological consequence dose calculation for the control room and audited the applicant's dose calculations with the new passive control room air filtration line to the VES. Based on this review of the applicant's analyses, the staff verified that the control room habitability system and the technical support center design in the AP1000 DCD, Revision 17, as modified in RAI response dated May 24, 2010, meet the dose acceptance criteria specified in GDC 19 and NUREG-0800 Section 15.0.3.

The staff previously stated that upon completion of the review and acceptance of this new passive control room air filtration line to the VES, the staff will complete its independent radiological consequence dose calculations for the control room to verify that the control room habitability system and the technical support center (TSC) designed in the AP1000 DCD meet the dose acceptance criteria specified in GDC 19 and NUREG-0800 Section 15.0.3. This was Open Item OI-SRP15.3-1-RSAC-01. With completion of staff's review and an independent dose calculation as discussed above, Open Item OI-SRP15.3-1-RSAC-01 is closed.

Section 6.4 of this report gives the staff's evaluation and acceptance of the new passive control room air filtration line.

15.3.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed DCD changes provide reasonable assurance that the radiological consequences resulting from any of the DBAs will fall within the dose acceptance criteria specified in NUREG-0800 Section 15.0.3, 10 CFR 52.47(a)(2), GDC 19, and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 for the EAB, LPZ, and control room. The staff further concludes that Revision 17 of the DCD also provides reasonable assurance that the AP1000 TSC will meet radiological consequences criteria specified in NUREG-0737, "Clarification of TMI Action Plan Requirements," Supplement 1 and in NUREG-0800 Section 15.0.3.

16. TECHNICAL SPECIFICATIONS

16.1 Introduction

Chapter 16.0, “Technical Specifications,” of the AP1000 design control document (DCD), provides the AP1000 generic technical specifications (GTS) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, “Technical specifications.” Each operating license issued by the Commission is required to include technical specifications (TS) that set forth the safety limits (SLs), limiting safety system settings (LSSSs), limiting conditions for operation (LCOs), and other limitations on facility operation deemed necessary for the protection of public health and safety. 10 CFR 50.36(a)(2) requires, among other things, that each applicant for design certification (DC) include in its application proposed GTS for the portion of the plant that is within the scope of the DC application. For the AP1000 design, these are accepted as documented in 10 CFR Part 52, “License, certifications, and approvals for nuclear power plants,” Appendix D, “Design Certification Rule for the AP1000 Design,” and NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.”

10 CFR 52.47(a)(11), “Contents of applications; technical information,” and 10 CFR 52.79(a)(30), “Contents of applications; technical information in final safety analysis report,” states that a DC applicant and a combined license (COL) applicant respectively are to propose TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a, “Technical specifications on effluents from nuclear power reactors.” COL applicants that reference a certified design are to propose plant-specific technical specification (PTS), including the GTS approved during the DC review. The COL applicant may propose deviations from the certified generic TS prior to issuance of the COL by requesting an exemption from the associated 10 CFR Part 52 appendix that codifies the certified design. A holder of a COL may propose changes to the TS in accordance with 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” in order to adopt approved changes to the standard technical specification (STS) when such changes apply.

This safety evaluation report (SER) chapter documents the U.S. Nuclear Regulatory Commission (NRC) staff review of the amended GTS proposed by the DC applicant for the AP1000 design and their associated Bases. The review is for completeness and correctness in regard to NRC requirements and guidance, and for consistency with related portions of the DCD. The TS are derived from the analyses and evaluations in the DCD.

16.2 Summary

The AP1000 design employs passive safety-related systems that rely on gravity and natural processes, such as convection, evaporation, and condensation. The AP1000 GTS were modeled after Revision 2 of NUREG-1431, “Standard Technical Specifications: Westinghouse Plants.” In some cases, the applicant developed TS beyond those in the STS to account for the advanced passive design features of the AP1000. In many instances, the AP1000 system design functions are similar to those of operating pressurized-water reactors (PWRs), even though the components and systems are new. The amendment to the AP1000 DC affects the following sections of the AP1000 GTS and Bases:

- Section 1

- Section 2
- Sections 3.1 through 3.9
- Section 4
- Section 5

The AP1000 GTS include reviewer's notes stating conditions that a COL applicant (or licensee) must satisfy in order to complete a particular GTS provision (e.g., incorporation of an NRC-approved methodology into a plant's licensing basis, or a staff determination that a licensee's probabilistic risk assessment (PRA) program is of adequate quality).

In some instances, detailed design information, equipment selection, instrumentation settings, or other information needed to establish appropriate TS and Bases was not provided during the review of the AP1000 DC or the amendment to the AP1000 DC. This information is identified in Chapter 16 of the DCD and in the GTS and Bases and will be included in the PTS by the applicant for a COL. Locations for the addition of this information are signified in the GTS by square brackets [ ] to indicate that the COL applicant must provide plant-specific values or alternative text.

As parts of the amendment to the AP1000 DC, the applicant proposed to complete some of the bracketed COL information items. Technical report (TR)-74A (APP-GW-GLR-064), "AP1000 Generic Technical Specifications Completion," Revisions 0 and 1, were submitted to document these changes.

The remaining changes to the AP1000 GTS are either results of modifications to the plant equipment designs or are to resolve inconsistency between various TS requirements and their supporting information in the associated TS Bases. Revisions 0 and 1 to TR-74C (APP-GW-GLN-075), "AP1000 Generic Technical Specifications for Design Changes," were submitted to document these changes.

The applicant also submitted TR-134 (APP-GW-GLR-134), "AP1000 DCD Impacts to Support COLA Standardization," to document any supplemental changes to the AP1000 GTS that were not included in TR-74A or TR-74C.

This SER addresses changes to the AP1000 DCD, since Revision 15. These revisions were prepared using the guidance in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," and to the extent applicable for the DC, using regulatory guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants," as a guide for format and content.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC for this area of review.

TS: TS are provided in DCD Tier 2, Chapter 16.

16.3 Regulatory Basis

16.3.1 Regulatory Requirements

The relevant requirements of the NRC's regulations for this area of review, and the associated acceptance criteria, are given in Chapter 16 of NUREG-0800, and are summarized below.

Review interfaces with other NUREG-0800 sections can be found in Chapter 16 of NUREG-0800.

Section 182a of the Atomic Energy Act of 1954 (AEA), as amended, requires that applicants for nuclear power plant operating licenses will state:

> such technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TS. In doing so, the NRC placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration, 33 *Federal Register* (*FR)* 18610, "Technical Specifications for Facility Licenses; Safety Analysis Reports," (December 17, 1968), the NRC noted that applicants were expected to incorporate into their TS "...those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Accordingly, 10 CFR 50.36(c) requires that TS include: (1) SLs and LSSSs, (2) LCOs, (3) surveillance requirement (SRs), (4) design features, and (5) administrative controls.

10 CFR 50.36(c)(2)(ii) requires that an LCO be established in TS for each item meeting one or more of the following four criteria:

- Criterion 1 - Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).

- Criterion 2 - A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 3 - A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 4 - An SSC shown by operating experience or a probabilistic safety assessment to be significant to public health and safety.

In accordance with 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 17, "Electric Power Systems"; GDC 21, "Protection System Reliability and Testability"; GDC 34, "Residual Heat Removal"; GDC 35, "Emergency Core Cooling"; GDC 38, "Containment Heat Removal"; GDC 41, "Containment Atmosphere Cleanup"; and GDC 44, "Cooling Water," those SSCs shown to be significant to public health and safety need to have sufficient independence, redundancy, and testability to perform their safety functions assuming single failure.

10 CFR 50.36a, requires that TS include procedures for control of radioactive effluents.

10 CFR 52.47(a)(11) requires that a DC applicant propose TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a.

16.3.2 Regulatory Guidance

The relevant NRC requirements for TS and Bases reviews, and the associated acceptance criteria, are given in Chapter 16 of NUREG-0800. They are summarized below. Areas of review that interface with other NUREG-0800 sections can also be found in Chapter 16 of NUREG-0800.

For the reasons discussed in detail below, the acceptance criteria adequate to meet the above requirements are included in the STS documents. The STS for PWRs are in three NRC NUREGs. For each NUREG, Volume 1 includes the TS and Volume 2 includes the associated TS bases. The STS include bases for SLs, LSSSs, LCOs, and associated action and SRs. The NUREGs for the STS for PWRs are as follows:

- NUREG-1430, “Standard Technical Specifications Babcock and Wilcox Plants”
- NUREG-1431, “Standard Technical Specifications Westinghouse Plants”
- NUREG-1432, “Standard Technical Specifications Combustion Engineering Plants“

The STS reflect the results of a detailed review of the application of the Interim Policy Statement criteria to generic system functions, which were published in a “Split Report” issued to the nuclear steam supply system vendor owners groups in May 1988. The STS also reflect the results of extensive discussions concerning various drafts of STS so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in the NUREGs provide an abundance of information regarding the extent to which the STS present the requirements necessary for protecting public health and safety.

On July 22, 1993, the NRC issued its Final Policy Statement (58 FR 39132), expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the AEA and 10 CFR 50.36. In the final policy statement, the NRC described the safety benefits of the STS and encouraged licensees, to the extent applicable to use the STS for PTS amendments and for complete conversions to improved TS. Major revisions to the STS were published in 1995 (Revision 1), 2001 (Revision 2), and 2004 (Revision 3).

The format and content for GTS and Bases prepared for a DC should use STS and applicable Bases to the extent possible, notwithstanding design-specific characteristics. As is appropriate, deviation from the STS, as well as design-specific characteristics, should be technically justified by an applicant and reviewed in detail by the NRC prior to approval.

16.3.3 Other Guidance

The June 2005 “Writer’s Guide for Plant-Specific Improved Technical Specifications,” prepared by the Technical Specifications Task Force (TSTF), provides specific guidance for the preparation of PTS. The purpose of the guide is to provide guidance on the format and content of the improved TS and to promote consistency in content, format, and style.

Design/plant-specific risk insights were developed by the staff for use during the review of AP1000 applications and are provided in a risk insights report. The risk insights were developed using information from the AP1000 DCD and AP1000 PRA. The risk insights were used to identify areas that warranted more detailed review and to identify equipment and systems that met Criterion 4 in 10 CFR 50.36(c)(2)(ii).

16.3.4 Applicable Generic Communication

The following generic communications issued by the NRC are TS-related and require consideration when developing TS and associated bases:

- Generic letter (GL) 88-016, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 3, 1988

- GL 91-004, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," April 2, 1991

- GL 96-003, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996

- GL 03-001, "Control Room Habitability," June 12, 2003

- GL 06-001, "Steam Generator Tube Integrity and Associated Technical Specifications," January 20, 2006

The following NRC generic safety issues (GSI) are TS-related and require consideration when developing TS and associated Bases:

- GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System"
- GSI-120, "On-Line Testability of Protection Systems"

16.4 Evaluation

The staff reviewed and evaluated the GTS and Bases to verify their accuracy and completeness. The staff also reviewed the GTS to confirm the appropriateness of the restrictions imposed by the GTS to ensure that an operating AP1000 will operate within its SLs and LSSSs, as described in the final safety analysis report (FSAR). The GTS must ensure that a plant designed and constructed in accordance with the AP1000 design will be operated so as to maintain the validity of the analyses in the FSAR during the operating lifetime of the plant. In particular, the GTS must require an AP1000 licensee to take specified actions, up to and including shutting down the plant, if one or more SSCs are not functioning as designed, such that the plant may not respond as predicted in the FSAR, including the accident analyses in FSAR Tier 2, Chapter 15. In addition, the GTS must include provisions to govern every SSC that meets one or more of the four criteria in 10 CFR 50.36(c)(2)(ii).

As described in more detail below, the staff verified the adequacy of the GTS primarily by comparing them with the STS developed for the operating fleet of power reactors. The staff developed each of these sets of STS by generically applying the criteria of 10 CFR 50.36(c)(2)(ii) to the SSCs included in the respective designs. Whether any set of STS is adequate to govern the operation of a particular power reactor cannot be determined without an evaluation of the TS as applied to the SSCs of the particular plant, considering the design as a whole. Currently, 75 of the 104 units of the operating fleet of nuclear plants use the STS, in whole or in part; the majority of these units use the Westinghouse STS in NUREG-1431.

While the staff has not approved the STS on a generic basis, it has implicitly approved them on a case-by-case basis through staff review of license amendment requests in which licensees of currently operating reactors have proposed to incorporate STS provisions in the existing custom technical specification (CTS) in their operating licenses. Some amendments have involved adoption of applicable STS on an item-by-item basis, while others have involved entire conversions of a plant's CTS to improved TS incorporating most, if not all, of the STS applicable to the particular design involved. The staff has evaluated and confirmed the adequacy of the model STS to ensure that particular plant SSCs will be operated in accordance with the analyses in individual plant FSARs in the context of these amendment requests. In addition, licensees of currently existing plants have employed STS pursuant to amendment requests granted by the NRC to govern the operation of their plants, and the staff has not identified any adverse effect on plant safety due to the adoption of the STS. Accordingly, the STS can be used as a model for the GTS to govern the operation of SSCs to the extent the AP1000 SSCs are similar in design and function to those governed by the STS. Use of the STS as guidance for the evaluation of the GTS in this manner allows the staff to determine whether the operation of SSCs in accordance with the GTS will assure that the analyses in the FSAR for these SSCs remain valid during plant operation. In view of its 2 Loop PWR design and the similar functions of many of its SSCs to the SSCs of a Westinghouse 4 Loop design, the Westinghouse STS in NUREG-1431 can be applied as guidance in evaluating most of the AP1000 GTS.

The staff evaluated each of the changes in the respective TS sections listed below. The applicant committed to making the changes in the final version of the AP1000 DCD that are identified in the AP1000 DCD, Revision 17. In a subsequent revision (Revision 18) of the AP1000 DCD, the applicant included these changes. In the review of confirmatory items in Revision 18, some conforming changes to the TS were identified. In addition, a number of editorial corrections were found. This resulted in several DCD changes for Chapter 16 that are included in Revision 19.

The staff did not review sections of the AP1000 GTS and Bases that were unaffected by the changes proposed in the AP1000 DCD. The technical evaluation for the sections that were not affected by the amendment can be found in NUREG-1793.

16.4.1 Use and Application

Section 1.0 of the AP1000 GTS includes definitions of terms used in the context of plant TS, and examples to illustrate the applications of logical connectors, completion times for required actions, and frequencies for SRs. Changes to AP1000 GTS Section 1.0 are as follows:

- In TS Section 1.1, the applicant proposed changes to the definition of "SHUTDOWN MARGIN," which is used in conjunction with TS Sections 3.1.1, 3.1.4, 3.1.5 and 3.1.6, to clarify how the gray rod cluster assemblies (GRCA) will be accounted for in the calculation of SHUTDOWN MARGIN. In request for additional information (RAI)-SRP16-CTSB-01, the staff requested additional details regarding this change. In its response, dated November 11, 2008, the applicant provided the requested information including a markup of changes to TS Section 1.1 in AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. Section 16.4.3.1 below provides the staff's evaluation of proposed changes to TS Section 3.1.

- In addition, the staff noted that an error in TS Section 1.4 had not been corrected in accordance with the NRC approved TSTF-485, "Correct Example 1.4-1," which corrects Example 1.4-1, Revision 0. RAI-SRP16-CTSB-02 was issued to the applicant for its correction. In its response, dated December 2, 2008, the applicant agreed to revise TS Section 1.4 in a future DCD revision. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the definitions for terms, logical connectors, conventions for completion times, and frequency requirements as provided in the Westinghouse STS. In addition, AP1000 GTS Section 1 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 1.0 of the AP1000 GTS is acceptable.

16.4.2 Safety Limits

Section 2.0 of the AP1000 DCD GTS and Bases include requirements for SLs, to ensure that the fuel design limits are not exceeded during steady state conditions, normal operational transients and anticipated operational occurrences.

The specifications provided in Section 2.0, which include the reactor core SLs and the reactor coolant system (RCS) pressure SL, are consistent with the STS and are found acceptable by the staff. Changes to AP1000 GTS Section 2.0 are as follows:

- In RAI-SRP16-CTSB-66, the staff asked the applicant to make an editorial change regarding an acronym in the bases of Section 2.1.1. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable section in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the SL information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 2 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 2.0 of the AP1000 GTS and Section B 2.0 of the AP1000 Bases are acceptable.

16.4.3 Limiting Condition for Operation and Surveillance Requirement Applicability

Section 3.0 of the AP1000 GTS and Bases includes general provisions regarding determination of equipment operability and performance of SRs in specific TS Section 3-series (i.e., TS Sections 3.1 through 3.9). There is no proposed change to AP1000 GTS Section 3.0.

16.4.3.1 Reactivity Control Systems

Section 3.1 of the AP1000 GTS and Bases includes requirements for the reactivity control systems, which are designed to reliably control reactivity changes and ensure that the capability to cool the core is maintained under postulated accident conditions.

The specifications provided in Section 3.1 consists of: Sections 3.1.1, “Shutdown Margin”; 3.1.2, “Core Reactivity”; 3.1.3, “Moderator Temperature Coefficient”; 3.1.4, “Rod Group Alignment Limits”; 3.1.5, “Shutdown Bank Insertion Limits”; 3.1.6, “Control Bank Insertion Limits”; 3.1.7, “Rod Position Indication”; 3.1.8, “Physics Tests Exceptions – Mode 2”; and 3.1.9, “Chemical and Volume Control System Demineralized Water Isolation Valves and Makeup Line Isolation Valves,” are consistent with the STS and are found acceptable by the staff. Changes to AP1000 GTS Section 3.1 are as follows:

- In RAI-SRP16-CTSB-34, the staff asked the applicant to clarify the mode of applicability for an SR in the bases of Section 3.1.1. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable section in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-67, the staff asked the applicant to make a minor editorial change regarding the title of an LCO in the Bases portions of Sections 3.1.4 and 3.1.8. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-05, the staff asked the applicant to clarify certain notes and their corresponding applicability modes in the specification and Bases portions of Sections 3.1.4, 3.1.5, and 3.1.6. In a letter dated November 19, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16CTSB-60, the staff asked the applicant to make an editorial change regarding required actions stated in the Bases of Section 3.1.7. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-43, the staff asked the applicant to make an editorial change regarding the required reactor power level stated in the specification and Bases portions of Section 3.1.8. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-20, the staff asked the applicant to make an editorial change regarding the correct revision year for a reference used in the Bases portion of Section 3.1.8. In a letter dated December 9, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the reactivity control systems information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.1 and its Bases do not include any “bracketed information” or “Reviewer’s Notes.” Therefore, the staff finds that Section 3.1 of the AP1000 GTS and Section B 3.1 of the AP1000 Bases are acceptable.

16.4.3.2 Power Distribution Limits

Section 3.2 of the AP1000 GTS and Bases includes requirements for the reactor core power distribution limits, which are designed to reliably control core thermal limits and core power distribution consistent with the design safety analysis. Changes to AP1000 GTS Section 3.2 are described as follows:

- The specifications provided in Section 3.2, which consists of Sections 3.2.1, “Heat Flux Hot Channel Factor”; 3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor”; 3.2.3, “Axial Flux Difference”; 3.2.4, “Quadrant Power Tilt Ratio”; and 3.2.5, “OPDMS-Monitored Parameters,” are consistent with the STS and are found acceptable by the staff.

- In RAI-SRP16-CTSB-68, the staff asked the applicant to make an editorial change regarding the documentation of the use of a reference in the Bases portion of Section 3.2.3. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-23, the staff asked the applicant to clarify the mode of applicability stated in the specification and Bases portions of Section 3.2.5. In a letter dated December 2, 2008, the applicant acknowledged the need for the change and included a mark-up of the applicable sections in the AP1000 DCD, Revision 17. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the power distribution limits information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.2 and its Bases do not include any “bracketed information” or “Reviewer’s Notes.” Therefore, the staff finds that Section 3.2 of the AP1000 GTS and Section B 3.2 of the AP1000 Bases are acceptable.

16.4.3.3 Instrumentation

Section 3.3 of the AP1000 GTS and Bases include requirements for the instrumentation systems that display information required to protect against violating the core fuel design limits and RCS, and to mitigate accidents. Changes to AP1000 GTS Section 3.3 are described as follows:

- Section 3.3, “Instrumentation,” of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 16, and TR-134, Revisions 0 through 5, the applicant made minor editorial changes and updated technical information. The applicant justified other editorial changes in TR-74C, Revision 0. RAI-SRP16-CTSB-69 was submitted to correct editorial errors in these changes. In the AP1000 DCD, Revision 17, the applicant corrected the editorial and typographical errors.

- The applicant removed the brackets [ ] around the completion times in Sections 3.3.1 and 3.3.2., and restored the 92-day frequency to SR 3.3.1.6 and SR 3.3.2.5. The applicant documented the basis for these changes in TR-74, APP-GW-GLR-064, Revision 1, April 13, 2007, “AP1000 Generic Technical Specifications Completion Update on Open Items,” and APP-GW-GSC-020, Revision 0, March 17, 2008, “AP1000 Protection and Safety Monitoring System Technical Specification Completion Time and Surveillance Frequency Justification.” The applicant has incorporated these changes in the AP1000 DCD, Revision 17. The applicant revised the SR completion times to be consistent with APP-GW-GSC-020.

- The applicant stated that ALL values specified for Trip Setpoints and Allowable Values, in Tables 3.3.1-1 and 3.3.2-1, must be confirmed following the completion of the plant-specific setpoint study. After selection of specific instrumentation, the Trip Setpoints can be calculated using the setpoint methodology described in Westinghouse Commercial Atomic Power (WCAP)-16361, APP-PMS-JEP-001, Revision 0, May 2006, “Westinghouse Setpoint Methodology for Protection Systems – AP1000.” In the AP1000 DCD, Revision 17, the applicant has removed all bracketed items for Trip Setpoints and Allowable Values in the tables, but includes a Reviewer’s Note to direct the COL applicant to use the approved methodology to calculate these values. A discussion of the acceptability of the AP1000 Setpoint Control Program (SCP), used to calculate setpoint values with this methodology, is included in Section 16.4.5 of this report.

- In TS 3.3.1, Table 3.3.1-1, equations for overtemperature ΔT (Note 1) and overpower ΔT (Note 2) are provided. The staff previously requested, in RAI-SRP16-CTSB-42, that the applicant provide the technical bases and derivation of the revised overtemperature ΔT and overpower ΔT reactor trip setpoint equations presented in Revision 16, and provide a reference to a document approved by the staff for the basis of the revised equations, or submit the basis for the revised equations to the staff for further review. The response provided for RAI-SRP16-CTSB-42 did not fully address the staff’s request. WCAP-8745-P-A, previously reviewed and approved by the staff, provided the bases for the overtemperature ΔT and overpower ΔT setpoint equations presented in Revision 15 of the DCD. The revised equations presented in the DCD Revision 16 for these reactor trip functions differ from those previously submitted in Revision 15 of DCD Section 7.2.1.1.3 and TS Table 3.3.1-1, Note 1.

  Based on this, the staff believed that the applicant should document the bases for the revised equations; the bases for development of the tables of allowable core thermal power as a function of core inlet temperature at various pressures for the overtemperature ΔT trip equation, the bases for the determination of the preset bias K4 in the overpower ΔT trip equation, and the bases for the constants and bracketed values that appear in the revised equations presented in Revision 16. The staff reviewed TR APP-GW-GLR-137, Revision 0, “Bases of Digital Overpower and Overtemperature

Delta-T (OPΔT/OTΔT) Reactor Trips," submitted by the applicant, in WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Over Temperature ΔT Trip Functions," September 1986 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML073521507). The content of this review is found in Chapter 7, Appendix 7.A of this report. The applicant described and clarified these items in more detail, as well as provided commitments to update references in the report and add this additional information. The staff finds these acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-44, the staff requested clarification/consistency of Function 6 (overtemperature ΔT) and Function 7 (overpower ΔT) "required channel" column in Table 3.3-1, "Reactor Trip System Instrumentation." The applicant added "4 (2/loop)" in the required channel column for clarification. This has been reviewed and accepted by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-45, the staff requested clarification/consistency of Function 12 (reactor coolant pump [RCP] speed-low) "required channel" column in Table 3.3-1, "Reactor Trip System Instrumentation." The applicant added "4 (1/pump)" in the required channel column for clarification. This has been reviewed and accepted by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-52, the staff requested resolution of conflicting information for the required minimum number of core exit thermocouples per core quadrant. The conflict was between Note (b) in Table 3.3.3-1, "Post Accident Monitoring," and DCD Table 7.5-1, "Instrumentation and Controls," Sheet 2. The applicant changed the number of instruments required from "2 quadrants" to "2 quadrants per Division" in Table 7.5-1, Sheet 2. This has been reviewed and accepted by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the instrumentation information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.3 and its Bases include "bracketed information." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 3.3 of the AP1000 GTS and Section B 3.3 of the AP1000 Bases are acceptable.

16.4.3.4 Reactor Coolant System

Section 3.4 of the AP1000 GTS and Bases include requirements for various RCS parameters (i.e., pressure, temperature, flow, etc.) and subsystems (i.e., RCS loops, pressurizer, low-temperature overpressure protection (LTOP), etc.) to ensure the fuel integrity and the RCPB integrity are preserved during all modes of plant operation. Changes to AP1000 GTS Section 3.4 are described as follows:

- In TS 3.4.1, the applicant proposed to use the preliminary bracketed value of 1.41 × 10^6 Liters per minute (Lpm) (301,670 gallons per minute (gpm)), specified in LCO 3.4.1.c for the minimum RCS total flow rate, as a final value based on latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. The staff finds this final value acceptable since it is consistent with supporting information provided in the TS Bases B 3.4.1 and relevant information described in AP1000 DCD Sections 4.4 (Table 4.4-1) and 15.0 (Table 15.0-3).

- The applicant also proposed to change requirements specified in SR 3.4.1.4 for monitoring RCS flow, to reflect an alternate testing method to the precision heat balance (an NRC-accepted method). In RAI-SRP16-CTSB-25, the staff asked the applicant to provide justification for the change. In its December 2, 2008, response, the applicant provided additional details about the basis for the alternate method and also stated the following:

  > The intent of the proposed Section 3.4.1 is to permit either method, whichever is demonstrated to provide less measurement uncertainty....The total uncertainty in measuring flow will depend upon analysis of the baseline flow measurements and the accuracy of the devices used to periodically measure dP caused by RCS flow. If the total uncertainty is not shown to be less than for the precision heat balance plus Delta-T method, then the alternate method would not be used.

  The applicant also indicated that no change to the AP1000 DCD or the TS 3.4.1 and associated bases is required.

  In reviewing this response, the staff noted that the alternate testing method using elbow tabs had been approved for use at the South Texas Project Electric Generating Station. A review of the current South Texas Project TS found the following descriptions for the affected SRs:

  > SR 4.2.5.2 The RCS flow rate indicators shall be subjected to a channel calibration at least once per 18 months.
  >
  > SR 4.2.5.3 The RCS total flow rate shall be determined by precision heat balance or elbow tab dP measurements at least once per 18 months.

  Based on the above, the staff believed a revision to the SR 3.4.1.4 and TS Bases 3.4.1 was needed to incorporate additional details regarding the choice of a testing method that produces better uncertainty analysis results, including a new SR for a channel calibration of the RCS flow rate indicators. In its response, dated August 20, 2009, the applicant proposed to: (1) add a new SR for a channel calibration for RCS flow indicators at the main control room board; and (2) revise SR 3.4.1.4 and TS Bases B 3.4.1 to incorporate a discussion of uncertainty analyses related to the use of elbow tabs as an alternate method for RCS flow verification. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.2, the applicant proposed to use the preliminary bracketed value of 288 °Celsius (C) (551 °Fahrenheit (F)), for the minimum RCS cold-leg temperature for criticality, as a final value based on historical relationships between the no-load operating temperature (292 °C (557 °F)), the minimum temperature for criticality (288 °C (551 °F)), and the limit for Mode 2 physics testing (283 °C (541 °F)). The staff finds this final value acceptable since it is consistent with supporting information provided in the TS Bases B 3.4.2 and relevant information described in AP1000 DCD, Sections 5.4 and 15.0.3.

- In TS 3.4.4, the applicant proposed to replace the preliminary bracketed values of 135 °C (275 °F) with a new final value of 93 °C (200 °F) and to use the preliminary bracketed value of 10 °C (50 °F) as a final value, regarding temperature requirements for the primary coolant and the secondary-side water as listed in Note 2 of LCO 3.4.4. In addition, the applicant proposed to add an extra precautionary note regarding restrictive plant conditions before starting an RCP for the reactor vessel LTOP. In RAI-SRP16-CTSB-55, the applicant was asked to provide clarification of the selected value of 93 °C (200 °F). This value 93 °C (200 °F) is not consistent with the one listed in the Westinghouse STS 135 °C (275 °F). In the March 23, 2009, response letter, the applicant proposed a further change from 93.3 °C ("200 degree F") to 177 °C ("350 degree F") based on an updated LTOP analysis, which now credits a technical design difference for AP1000 related to the variable-speed RCP start-up design limitations (e.g., RCPs are required to be started at a relatively slow pump speed and they are unable to start at full speed) not recognized in the Westinghouse STS (NUREG-1431). The staff finds the changes and justification as provided in this latest response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Although no change was proposed to TS 3.4.6 as part of the AP1000 DC amendment application, the staff noted inconsistencies between SR 3.4.6.1 requirements and supporting information in Bases B 3.4.6, regarding lift setpoints for pressurizer safety valves. In RAI-SRP16-CTSB-08, the staff asked the applicant to address these inconsistencies. In its response dated December 17, 2008, the applicant proposed to revise the Bases for SR 3.4.6.1 to indicate +/- 1 percent OPERABLE range for the valve lift settings, to be consistent with SR 3.4.6.1 and with the tolerance established in WCAP-16779, "AP1000 Overpressure Protection Report," April 2007. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.7, the applicant proposed to delete LCO 3.4.7.d for the total primary-to-secondary leakage through both steam generators (SGs) because it is redundant to LCO 3.4.7.e for primary to secondary leakage through any one SG. In addition, changes were proposed to SR 3.4.7.2 to reflect the implementation of a new SG program to maintain the SG tube integrity. The staff agreed with the applicant's position on deleting LCO 3.4.7.d and finds the proposed changes to SR 3.4.7.2 acceptable since they are consistent with other requirements in the AP1000 GTS (GTS 3.4.18, GTS 5.5.4, and GTS 5.6.8).

- In TS 3.4.8, the applicant proposed to add a missing clarifier to the applicability statement that allows stopping all RCPs without having to enter an action statement. The staff finds this change acceptable since the added special plant condition is consistent with remaining TS 3.4.8 requirements.

- The applicant also proposed to replace the preliminary bracketed value of 37,785 Lpm (10,000 gpm) for the minimum RCS flow with a final value of 11,356 Lpm (3,000 gpm). Conforming changes were proposed in SR 3.4.8.1 and related information in the TS Bases to match the new minimum flow value (e.g., the minimum pump speed setting of 25 percent was replaced with a new value of 10 percent). The applicant cited the NRC-accepted response to RAI 440.106 during the Revision 14 AP1000 DC review as justification for the proposed flow reduction. In RAI-SRP16-CTSB-62, the staff asked the applicant to provide additional details to support these changes. In its response dated December 17, 2008, the applicant reiterated information that was provided in the response to RAI 440.106 but also stated the following:

  > AP1000 RCS flow calculations show that the expected RCS flow with a single reactor coolant pump (RCP) operating at its lowest allowable operating speed is approximately 64,352 Lpm (17,000 gpm). The associated reactor vessel flow is approximately 41,640 Lpm (11,000 gpm). This is well above the 11,356 Lpm (3,000 gpm) flow mixing requirement from the LOFT testing, and also above the preliminary bracketed value of 37,854 Lpm (10,000 gpm).

  The staff noted that the new proposed value of 10 percent for the pump minimum speed setting in SR 3.4.8.1, corresponding approximately to a calculated flow of 29,810 Lpm (7,875 gpm), appears to be inconsistent with the lowest allowable operating speed stated above. In its response, dated July 15, 2009, the applicant stated that the mentioned lowest allowable operating speed is for equipment protection while the minimum speed specified in TS 3.4.8 is to reflect the assumed RCS flow through the core in shutdown event analyses. The TS limit will be satisfied if the operating limit is maintained in accordance with plant equipment operating procedures. The staff finds this response acceptable and RAI-SRP16-CTSB-62 is closed.

- Although no change was proposed to TS 3.4.11/12 as part of the AP1000 DC amendment application, the staff noted that the scope of Condition A was not clearly defined. In RAI-SRP16-CTSB-07, the staff asked the applicant to explain the difference in scope of inoperable equipment involved between TS 3.4.11/3.4.12 Condition A, which states “One required flow path inoperable,” and Condition B, which states, “One required stage 1 ADS flow path inoperable AND Either one required stage 2 or stage 3 ADS flow path inoperable.”

  In its October 27, 2008, response letter, the applicant stated the following:

  > As described in the 3.4.11 and 3.4.12 Bases, Conditions A and B cover two different combinations of ADS flow path inoperabilities.. Separate Conditions are specified, since both Conditions A and B may be entered at the same time. The inoperabilities covered by the two Conditions are permissible at the same time, since the safety function can be accomplished by the remaining seven ADS flow paths without a single failure. The loss of capacity while in Conditions A and B is equivalent to a single failure of the power to the valves in one division, as considered in the accident analyses.

  The applicant further stated “the LCO 3.4.11 and LCO 3.4.12 and associated Bases are technically correct, as-is. However, to clarify the system status while in both

Conditions A and B the following statement is added in each of the Bases at the beginning of the Actions sections:

> The loss of automatic depressurization system (ADS) capacity, if both Conditions A and B are entered at the same time, is equivalent to a single failure of the power to the valves in one division, as considered in the accident analyses."

Based on this response and considering the four-stage ADS design, the staff believed that additional changes were required for Condition A to explicitly list Stage 4 ADS flow path in its scope and to clearly indicate the difference between Conditions A and B. In its July 15, 2009, response, the applicant proposed to revise all Action statements in TS 3.4.11 and TS 3.4.12, and the associated TS bases to clearly define the scope of inoperable ADS valves for each LCO condition and to assign a completion time consistent with guidance in the STS for cases with the same remaining operable ADS valves. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.14, the applicant proposed to replace the preliminary bracketed value of 152.4 square centimeters (cm^2) (9.3 square inches (in^2)) for the minimum RCS vent area with a final value of 68 cm^2 (4.15 in^2). In RAI-SRP16-CTSB-35, the staff asked the applicant to provide a justification for the change. In its response dated December 12, 2008, the applicant stated that the change is a result of the final design of the normal residual heat removal system (RNS) suction relief valve with its inlet changed from 10.16 centimeters (cm) (4 inches(in)) to 7.62 cm (3 in). The staff finds the stated justification acceptable since either the RNS suction relief valve or a depressurized RCS with a vent area is considered an acceptable means for providing LTOP. The staff considers RAI-SRP16-CTSB-35 closed.

- In addition, in RAI-SRP16-CTSB-54, the staff asked the applicant to address inconsistencies in the TS bases B 3.4.14 regarding discussion on restarting of one RCP as a heat input event. In its March 23, 2009, response, the applicant proposed to revise LCO 3.4.14 and its associated TS bases to make the descriptions of LCO 3.4.14, Notes 1 and 2 consistent with those specified in LCO 3.4.4 and LCO 3.4.8 for the same situation. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 3.4.15, the applicant proposed to use the preliminary bracketed value of 15,272 kilopascal (kPa) (2,215 pounds per square inch gauge (psig)) and 15,549 kPa (2,255 psig), for the range of RCS pressure during performance of SR 3.4.15.1 to verify leakage through each RCS pressure isolation valve, as final values based on the nominal RCS pressure design of AP1000 and the requirements for test pressures identified in American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code ISTC-3630(b). The staff finds the stated reason for the final selected values acceptable.

- At the end of TS Section 3.4, the applicant proposed to add a new TS 3.4.18, "Steam Generator Tube Integrity," to reflect implementation of the NRC-approved TSTF-449, "Steam Generator Tube Integrity," Revision 3. The staff finds the proposed addition of TS 3.4.18 acceptable since implementing TSTF-449 is one acceptable option for addressing the safety issues identified in GL 06-001.

The applicant adhered to the RCS information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 3.4 and its Bases do not include any "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.4 of the AP1000 GTS and Section B 3.4 of the AP1000 Bases are acceptable.

16.4.3.5 Emergency Core Cooling Systems

Section 3.5 of the PTS and Bases includes requirements for the safety-related passive core cooling system (PXS), which is designed to perform emergency core decay heat removal, RCS emergency makeup and boration, and safety injection. Changes to AP1000 GTS Section 3.5 are described as follows:

- In TS Section 3.5.2, the applicant proposed to use the preliminary bracketed value of 0.0057 cubic meters (m^3) (0.2 cubic feet (ft^3)), for the maximum allowable volume of noncondensable gases in each of the core makeup tanks' inlet piping, as a final value based on latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. The staff finds this final value acceptable since it is consistent with related information described in AP1000 DCD Section 6.3.

- In TS Section 3.5.4, the applicant proposed to replace the preliminary bracketed value of 0.011 m^3 (0.4 ft^3), for the maximum allowable volume of noncondensable gases in the passive residual heat removal heat exchanger inlet piping, with a new final value of 0.025 m^3 (0.9 ft^3). In RAI-SRP16-CTSB-36, the staff asked the applicant to provide justification for the change. In its response dated December 12, 2008, the applicant stated that the value of 0.025 m^3 (0.9 ft^3) reflects the correct design value based on the final location for the alarm limit switch installed in the high-point pipe stub section. The staff finds this final value acceptable based on verification that a physical change was made in AP1000 DCD Section 6.3, regarding an increase in pipe size at the level switch location from 0.305 meters (m) to 0.355 m (12 in to 14 in). Therefore, the staff considers RAI-SRP16-CTSB-36 closed.

- In TS Sections 3.5.6 and 3.5.8, the applicant proposed to replace the preliminary bracketed value of 2091 m^3 (73,900 ft^3), for the minimum volume of borated water in the in-containment refueling water storage tank (IRWST), with a new final value of 2069 m^3 (73,100 ft^3). In RAI-SRP16-CTSB-37, the staff asked the applicant to provide justification for the change. In its response dated December 12, 2008, the applicant stated the following:

    > The bracketed volume of 2091 m^3 (73,900 ft^3) represented a preliminary estimate of the minimum design basis IRWST water volume.
    >
    > The un-bracketed value of 2069 m^3 (73,100 ft^3) was updated based on evolving IRWST design details, is consistent with the updated IRWST volume provided in DCD Table 6.3-2 (Sheet 2), and reflects a more

> conservative water volume that was appropriately used in safety analyses."

The staff finds the stated reason acceptable and considers RAI-SRP16-CTSB-37 closed.

The applicant adhered to the ECCS information as provided in the Westinghouse STS. In addition, the AP1000 GTS, Section 3.5, and its Bases do not include "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.5 of the AP1000 GTS and Section B 3.5 of the AP1000 Bases are acceptable.

16.4.3.6 Containment Systems

Section 3.6 of the AP1000 DCD GTS and its Bases address requirements for the containment systems, which are designed to contain fission products that may exist in the containment atmosphere following accident conditions.

The specifications provided in Section 3.6 consist of: Sections 3.6.1, "Containment"; 3.6.2, "Containment Air Locks"; 3.6.3, "Containment Isolation Valves"; 3.6.4, "Containment Pressure"; 3.6.5, "Containment Air Temperature"; 3.6.6, "Passive Containment Cooling System (PCS) – Operating"; 3.6.7, "PCS Shutdown"; 3.6.8, "Containment Penetrations"; and 3.6.9, "pH Adjustment," are consistent with the STS and are found acceptable by the staff. Changes to AP1000 GTS Section 3.6 are as follows:

- In RAI-SRP16-CTSB-15, the staff asked the applicant to correct Bases B 3.6.6 to accurately reflect the action statements in TS 3.6.6. In a letter dated October 17, 2008, the applicant acknowledged the need for the change and included a markup of the applicable sections in AP1000 DCD, Revision 17. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-16, the staff asked the applicant to correct Bases B 3.6.7 to accurately reflect the action statements in TS 3.6.7. In a letter dated October 17, 2008, the applicant acknowledged the need for the change and included a markup of the applicable sections in AP1000 DCD, Revision 17. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16-CTSB-13, the staff asked the applicant to clarify Bases B 3.6.4 regarding maximum peak containment pressure. In a letter dated December 2, 2008, the applicant acknowledged the need for clarification and included a markup of the changes that will be incorporated. The staff finds this change acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In RAI-SRP16.1.1-SEB1-01, the staff asked the applicant to provide details on the equipment hatch and bolt design to ensure that the equipment hatch can be safely installed with four bolts to meet the containment closure requirements during Modes 5 and 6 (TS 3.6.8). In its response dated August 15, 2008, the applicant stated that design specification document APP-MV50-Z0-002, "Equipment Hatch Design

Certification Document," would provide final design information for the equipment hatch installation.

In an audit on November 30, 2009, the staff confirmed that design specification document APP-MV50-Z0-002, Revision 1, provides design criteria for equipment hatch bolts, such that its weight can be supported with only four bolts installed. A typical vendor report prepared for the AP1000 China Sanmen Unit 1 lower equipment hatch by IHI Corporation in Japan was also provided for the staff's review during this audit. The staff noted that this vendor report adequately addresses the applicable design criteria specified in APP-MV50-Z0-002. Although this vendor report was not prepared specifically for a new AP1000 plant to be built in the United States, the staff finds the applicant's approach to address this issue acceptable. The equipment hatch is listed as an ASME Code Section III component in AP1000 DCD Tier 1, Table 2.2.1-1, and as such, ITAAC Item 2.a in Table 2.2.1-3 is applicable to it. This ITAAC item calls for the existence of a design report for the as-built component received from the equipment supplier. The design document APP-MV50-Z0-002, together with the cited ITAAC item, will ensure equipment hatch components will be constructed and installed in accordance with design requirements. Therefore, the staff finds the response acceptable and RAI-SRP16.1.1-SEB1-01 is closed.

- In RAI-SRP16-CTSB-61, the staff asked the applicant to specify the sections of DCD Chapter 15 that support the specific accident discussed in the "Applicable Safety Analyses" section of TS Bases B 3.6.1, B 3.6.2, and B 3.6.3. In its November 19, 2008, response, the applicant stated that:

  > the level of detail provided by the B 3.6.1, B 3.6.2, and B 3.6.3 Bases references to Chapter 15 is consistent with that provided in the STS

  The applicant made no further change to the Bases. The staff found this reason unacceptable. The staff's concern was that DCD Chapter 15 is voluminous as it includes more than 600 pages. Without references to specific sections, validation of the information discussed in the affected TS bases would require significant effort and time from the plant operators who implement TS requirements and often refer to the TS bases for clarifications needed quickly. In its July 15, 2009, response, the applicant provided sufficient details to support its position that reference to specific sections of Chapter 15 is not helpful in these cases for the containment boundary as a physical barrier. The containment integrity and leak tightness are applicable to a wide range of accident scenarios described in Chapter 15. The staff finds this response acceptable and RAI-SRP16-CTSB-61 is closed.

- In RAI-SRP16-CTSB-33, the staff asked the applicant to provide the value of the minimum tri-sodium phosphate (TSP) manufactured density that is used to convert the required TSP amount from a mass number to a volume number. In its December 12, 2008, response, the applicant did not provide the requested information so that the staff could verify the accuracy and completeness of supporting information provided in TS Bases B 3.6.9. In its August 20, 2009, response, and the subsequent October 29, 2009, conference call, the applicant provided the requested information and agreed to revise TS Bases B 3.6.9 to include the new details. The staff finds this response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the containment information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 3.6 and its Bases include "bracketed information" and "Reviewer's Notes." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 3.6 of the AP1000 GTS and Section B 3.6 of the AP1000 Bases are acceptable.

16.4.3.7 Plant Systems

Section 3.7 of the PTS and Bases include requirements for various systems in the secondary side of the SGs (i.e., the main steam safety valves (MSSVs), the main steam isolation valves (MSIVs), the main feedwater isolation valves (MFIVs), etc.), the spent fuel pool water level and makeup systems, and the main control room habitability system. Changes to AP1000 GTS Section 3.7 are described as follows:

- In TS Section 3.7.1, "Main Steam Safety Valves (MSSVs)," the applicant proposed a slight increase in the relief capacity and the resulting relief setpoint for all but the first-to-open MSSV based on a minor change to the valve inlet piping to conform to ASME Code requirements. Also, the applicant replaced the bracketed values for the restriction on maximum allowable thermal power with inoperable MSSVs in Table 3.7.1-1 with new final values. The staff finds the final data in Table 3.7.1-1 acceptable since they were derived using methodology referenced in the Westinghouse STS, Revision 3.

- In addition, the applicant proposed to change the tolerance for the as-found relief setting for MSSVs in Table 3.7.1-2 from 1 percent to 3 percent. In RAI-SRP16-CTSB-11, the staff asked the applicant to provide justification for the change in Table 3.7.1-2. In its response dated December 17, 2008, the applicant proposed to change this tolerance back to the original value of 1 percent. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS Section 3.7.6, "Main Control Room Habitability System," the applicant proposed to use the preliminary bracketed value of 23,443 kPa (3,400 psig) for the required minimum pressure specified in SR 3.7.6.2, as a final value based on latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. The staff found the proposed final value acceptable since it is consistent with relevant information described in AP1000 DCD Section 6.4.2.

- In addition, the staff noted that the AP1000 GTS did not incorporate the NRC-approved TSTF-448, "Control Room Habitability," which was issued to address safety issues identified in GL 2003-001. In RAI-SRP16-CTSB-32, the staff asked the applicant to address these issues. In its response dated November 11, 2008, the applicant stated that it had added a new DCD Section 6.4.5.4, "Main Control Room Envelope Habitability," under Revision 16 to address GL 2003-001. This DCD section describes the periodic testing of the main control room envelope habitability during main control room emergency habitability system operation (pressurization mode) to measure the air inleakage in accordance with the American Society for Testing and Materials (ASTM) E741, "Testing of Palo Verde Units 1-3 Control Room Envelopes." The applicant concluded that this periodic testing commitment in DCD Section 6.4.5.4, combined with the existing LCO 3.7.6 requirements, adequately addresses the GL 2003-01 issues and provides requirements equivalent to those approved in TSTF-448. The applicant proposed no further changes to the AP1000 DCD or the AP1000 GTS. The staff disagreed with this conclusion. In the May 4, 2009, response letter to RAI-SRP6.4-SPCV-01, the applicant proposed to revise TS 3.7.6 and its associated bases to fully incorporate all TSTF-448 requirements. The staff finds this latest response acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. Also in response to RAI-SRP6.4-SPCV-06, the applicant proposed additional design changes to the control room habitability system. As a result, further changes are proposed to TS 3.7.6 and a new administrative control TS 5.5.13 is added to the AP1000 GTS for testing of the new passive filtration unit. Additional open items to TS 3.7.6 and TS 5.5.13 were identified and documented as part of the staff's evaluation of these design changes in SER Section 6.4. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS Section 3.7.9, “Fuel Storage Pool Makeup Water Source,” the applicant provided additional information to the precautionary Note 1 in LCO 3.7.9 for clarification. The staff finds the added text acceptable since it is consistent with relevant information described in AP1000 DCD Section 9.1.3.

- At the end of TS Section 3.7, the applicant proposed to add TS 3.7.11, “Fuel Storage Pool Boron Concentration,” and TS 3.7.12, “Spent Fuel Pool Storage,” to reflect the final design of the spent fuel storage racks. The staff finds the added TS requirements and associated information in the TS bases acceptable since they were formulated in accordance with guidance provided in the Westinghouse STS 3.7.16 and 3.7.17, respectively, and are consistent with relevant information in the AP1000 DCD, Section 9.1. Section 9.1 of this report presents a separate evaluation of the final design of the spent fuel storage racks.

The applicant adhered to the plant systems information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 3.7 and its Bases do not include “bracketed information” or “Reviewer’s Notes.” Therefore, the staff finds that Section 3.7 of the AP1000 GTS and Section B 3.7 of the AP1000 Bases are acceptable.

16.4.3.8 Electrical Power Systems

Section 3.8 of the AP1000 GTS and Bases include requirements for the plant electrical systems that provide redundant, diverse and dependable power sources for all plant operating conditions. In the event of a total loss of offsite power, onsite diesel generators and batteries are provided to supply electrical power equipment necessary for the safe shutdown of the plant. Changes to AP1000 GTS Section 3.8 are as follows:

- Section 3.8, “Electrical Power Systems,” of the AP1000 DCD, Revision 15, was approved by the staff in the certified design. In the AP1000 DCD, Revision 16, and TR-134, Revisions 0 through 5, the applicant made minor editorial changes and updated technical information. The staff finds these editorial changes acceptable.

- In the AP1000 DCD, Revision 17, the applicant replaced all preliminary information in the brackets with the final information. The applicant documented the basis for these changes in TR-74, Revision 1. The staff finds these changes acceptable due to the justifications found in the reference report.

- The applicant initially proposed retaining brackets [ ] around all preliminary AP1000 DCD values associated with the battery float current. COL applicants referencing the AP1000 DCD would replace preliminary information, provided in brackets [ ], with final plant specific values. In the AP1000 DCD, Revision 17, the applicant replaced all preliminary information in the brackets with the final information. The staff finds this acceptable since it is consistent with the guidance provided in the Institute of Electrical and Electronic Engineers (IEEE) references.

- The applicant inadvertently omitted the “7 days” completion time in TS Section 3.8.1 B.3 and has added it into AP1000 DCD, Revision 17, and Revision 4 of TR-134. The staff finds this editorial change acceptable.

The applicant adhered to the electrical power systems information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.8 and its Bases do not include “bracketed information” or “Reviewer’s Notes.” Therefore, the staff finds that Section 3.8 of the AP1000 GTS and Section B 3.8 of the AP1000 Bases are acceptable.

16.4.3.9 Refueling Operations

Section 3.9 of the AP1000 GTS and Bases includes requirement for boron concentration, unborated water sources, nuclear instrumentation, containment penetrations, and water inventory in the refueling pool during Mode 6. Changes to AP1000 GTS Section 3.9 include the following:

- In TS Section 3.9.5, “Containment Penetrations,” the applicant proposed to use the preliminary bracketed information as a final value for the required number of bolts (four) to keep the equipment hatch in place to meet the containment closure requirements during movement of irradiated fuel assemblies within the containment, based on the latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. In RAI-SRP16.1.1-SEB1-01, the staff asked the applicant to provide additional details on the bolt design to ensure the safe installation of the equipment hatch with only four bolts. In its response dated August 15, 2008, the applicant stated that design specification document APP-MV50-Z0-002 would provide final design information for the equipment hatch installation. The discussion and closure of RAI-SRP16.1.1-SEB1-01 is found above in Section 16.4.3.6.

- Also, in TS Sections 3.9.5, “Containment Penetrations”; and 3.9.6, “Containment Air Filtration System (VFS),” the applicant proposed to use the preliminary bracketed value of -0.0311 kPa (-0.125 inches water gauge) relative to outside atmospheric pressure for VFS subsystem testing in SR 3.9.5.3 and SR 3.9.6.3. The applicant proposed using the preliminary value as a final value based on the latest system design specifications, approved engineering calculation notes, and/or verified analysis input assumptions. In RAI-SRP16-CTSB-59, the staff asked the applicant to explain the basis for the selected value. In its response dated August 15, 2008, the applicant stated the following:

> This pressure was chosen based on ASHRAE Applications, which recommends at least 0.0124 kPa to 0.0149 kPa (0.05 to 0.06 inches of water) across boundaries when exfiltration or infiltration is minimized. Conservatively, Westinghouse chose a higher pressure difference of 0.0311 kPa (0.125 inches of water).

The staff finds the stated reason acceptable since the selected value is more conservative than the value used in normal industry practices. Therefore, the staff considers RAI-SRP16-CTSB-59 closed.

- In TS Section 3.9.7, "Decay Time," the applicant proposed to change the minimum decay time of 100 hours to 48 hours to make it consistent with the analysis of the fuel handling accident as described in AP1000 DCD Section 15.7.4. The staff finds this change acceptable for the stated reason.

The applicant adhered to the refueling operations information as provided in the Westinghouse STS. In addition, AP1000 GTS Section 3.9 and its Bases do not include "bracketed information" or "Reviewer's Notes." Therefore, the staff finds that Section 3.9 of the AP1000 GTS and Section B 3.9 of the AP1000 Bases are acceptable.

16.4.4 Design Features

Section 4.0 of the AP1000 GTS includes other design features not covered in TS Section 3-series, such as the site location, the site maps, and other information related to core design and fuel storage design. Changes to AP1000 GTS Section 4.0 are as follows:

- In TS Section 4.3, "Fuel Storage," the applicant proposed various changes to the description of the fuel storage area to reflect the final design for new and spent fuel storage racks and an increase of the maximum capacity of the spent fuel storage racks from 616 to 889 fuel assemblies. Evaluation of the final design modification is provided separately in Section 9.1 of this report. Furthermore, in RAI-SRP16-CTSB-38 and RAI-SRP16-CTSB-39, the applicant was asked to address inconsistencies between the information provided in TS Section 4.3 and DCD Section 9.1. In its response dated December 2, 2008, the applicant proposed revisions to TS Section 4.3 and DCD Section 9.1 to revolve these inconsistencies. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the design features information as provided in the Westinghouse STS, with differences to reflect AP1000 unique design features. With respect to AP1000 unique design features, the GTS are sufficient to assure operation of these features within the bounds of the safety analysis. In addition, AP1000 GTS Section 4.0 includes "bracketed information" and "Reviewer's Notes." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 4.0 of the AP1000 GTS is acceptable.

16.4.5 Administrative Controls

Section 5.0 of the AP1000 GTS includes provisions, which address various administrative controls related to plant key personnel responsibilities, plant procedures, special programs and reports, etc., to ensure the plant is safely operated. Changes to AP1000 GTS Section 5.0 are described as follows:

- In TS Section 5.4, "Procedures," the applicant proposed to adopt GL 1982-33, "Supplement 1 to NUREG-0737—Emergency Response Capabilities," dated December 17, 1982, as guidance to be used in the development of the plant emergency operating procedures. This is consistent with the STS and acceptable to the staff.

- In TS 5.5, "Programs and Manuals," and in TS 5.6, "Reporting Requirements," the applicant proposed changes to TS 5.5.4, "Steam Generator Program," and to TS 5.6.8, "Steam Generator Tube Inspection Report," to reflect the implementation of the NRC-approved TSTF-449, Revision 4. The staff finds these changes acceptable since implementing TSTF-449 is one acceptable option for addressing safety issues identified in GL 2006-001. However, since TSTF-449 was prepared to address issues involving SG replacements at current operating plants, in RAI-SRP16-CTSB-76, the staff asked the applicant to make one minor adjustment to its proposed changes in TS 5.5.4 to also accommodate SG initial installations at new nuclear power plants regarding the 100-percent tube inspection during the first refueling outage. In its response dated December 2, 2008, the applicant agreed to make the suggested adjustment in a future DCD revision. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- In TS 5.5.8, the applicant proposed to use the preliminary bracketed numerical values as final values for acceptance criteria used in various tests on the containment air locks. The staff finds these final selected values acceptable since they are consistent with recommendations provided in the Westinghouse STS.

- In connection to TS 3.7.6 regarding implementation of TSTF-448, Revision 3, to address safety issues identified in GL 2003-01, in RAI-SRP16-CTSB-32, the staff asked the applicant to include the description of the Control Room Habitability Program into AP1000 GTS Section 5.5. Discussions and closure of RAI-SRP16-CTSB-32 are addressed in Section 16.4.3.7 above.

- Also, in TS 5.5.11, "Battery Monitoring and Maintenance Program," the applicant proposed to adopt the preliminary bracketed texts that are applicable to "vented lead-acid" batteries, as the final texts based on latest system design specifications. The staff finds this acceptable since it is consistent with recommendations provided in the Westinghouse STS.

- As stated in Section 16.4.3.3 above, in response to RAI-SRP16.3-CTSB-SCP-1, the applicant stated that all values specified for trip setpoints and allowable values in Tables 3.3.1-1 and 3.3.2-1 will be determined via a SCP specified in Section 5.5.21, "Administrative Controls." This is in accordance with DC/COL-ISG-8, "Technical Specification Information that Combined License Applicants Must Provide in Combined License Application," in determining instrumentation trip setpoints and allowable values in TS. After selection of specific instrumentation, the trip setpoints can be calculated using the setpoint methodology specified in the SCP; in WCAP-16361, Revision 0. The staff finds the applicant's response to RAI-SRP16.3-CTSB-SCP-1 acceptable. The staff received the applicant's proposed SCP in an RAI response dated February 19, 2010. Initially, the staff found the proposed SCP unacceptable, and communicated their concerns to the applicant. On May 6, 2010, a revised program was submitted to the staff for review. The staff reviewed the revised SCP and found it to be consistent with the recommendations provided in DC/COL-ISG-8. Based on the comprehensive nature of the SCP, the staff believes a COL applicant can calculate all values necessary to complete information found in Section 3.3 of the TS. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant adhered to the administrative controls information as provided in the Westinghouse STS, except as noted above. In addition, AP1000 GTS Section 5.0 includes "bracketed information" and "Reviewer's Notes." The staff reviewed each piece of "bracketed information" to understand its intent and to determine whether each was site-specific and appropriately deferred to applicants for construction permits or COLs that reference the AP1000 GTS. The staff concluded that each such item was indeed plant- or site-specific. Therefore, the staff finds that Section 5.0 of the AP1000 GTS is acceptable.

16.5 Conclusion

The staff concludes that the changes to the AP1000 GTS and Bases include design-specific parameters and additional TS requirements considered appropriate by the staff. In addition, the staff has compared the additional TS requirements to the relevant NRC regulations, acceptance criteria defined in NUREG-0800, Section 16.0, and other guidance and concludes that the application is in compliance with NRC regulations.

For the reasons set forth above, the staff finds the changes to AP1000 DCD Chapter 16, GTS and Bases, are acceptable and satisfy the requirements of 10 CFR 50.36, 10 CFR 50.36a, and 10 CFR 52.47(a)(11).

17. QUALITY ASSURANCE

17.3 Quality Assurance During the Design Phase

17.3.1 Introduction

Revision 17 of the AP1000 design control document (DCD) proposes to implement the Westinghouse Quality Management System (QMS), Revision 5, for the AP1000 project. The Nuclear Regulatory Commission (NRC) approved Revision 5 to the Westinghouse QMS in a letter dated September 13, 2002. Revision 0 of technical report (TR)-109, APP-GW-GLR-109, "DCD Revision to Incorporate ASME NQA-1-1994 for AP1000," presents the technical justification for implementing the QMS for the AP1000 project. TR-109 justifies standard changes to the AP1000 DCD Sections 3.8, 5.2, 5.4, 17.3, and Appendix 1A. TR-109 also incorporates the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA) Standard NQA-1–1994, "Quality Assurance Requirements for Nuclear Facility Applications," into the DCD for AP1000.

17.3.2 Evaluation

NRC regulatory guide (RG) 1.28, Revision 3, "Quality Assurance Program Requirements (Design and Construction)," issued August 1985, endorsed NQA-1-1983 as an acceptable method for complying with the provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," with regard to the requisite quality assurance (QA) program for the design and construction phases of nuclear power plants.

In Revision 15 of the AP1000 DCD, the applicant committed to the guidance of ASME NQA-1b-1991. As stated in TR-109, the proposed change would revise the version of NQA-1 referenced and committed to in the AP1000 DCD from NQA-1b-1991 to NQA-1-1994. The NRC staff compared the two versions of NQA-1 and found that the 1994 Edition differed from the 1991 Edition primarily in format. NQA-1-1994 consolidates NQA-1 and NQA-2 into a single multipart document and restructures its format to facilitate use of various parts of the standard. The basic requirements of the 1991 Edition were substantially unchanged. Additionally, the applicant evaluated the changes from NQA-1b-1991 to NQA-1-1994. As documented in TR-109, these changes include: (1) reordering of definitions; (2) addition of functions for which personnel need to be qualified; (3) addition of "siting" to the list of activities affecting quality and the list of quality functions to be audited; (4) addition of a paragraph for inspection requirements; and (5) removal of the allowance related to obsolete drawings. The applicant determined that these changes did not represent a reduction in commitment and did not impact the AP1000 design.

In a letter dated June 6, 2008, the applicant responded to the NRC's request for additional information (RAI)-SRP17.3-CQVP-01 regarding missing changes to Revision 16 of the AP1000 DCD associated with TR-109. Specifically, the NRC staff noted that DCD Sections 3.8.3.6, 5.2.3.4.1, and 5.2.4.6 were not modified as described in TR-109, nor did Revision 4 of TR-134, APP-GW-GLR-134, "AP1000 DCD Impacts to Support COLA Standardization," capture the changes. In its response, the applicant stated that Revision 5 of TR-134 would capture the supportive changes described in TR-109. The NRC staff confirmed that TR-134, Revision 5, captures the proposed changes. In addition, the NRC staff confirmed that the changes described in TR-109 and Revision 5 of TR-134 are included in Revision 17 of the AP1000 DCD.

As stated above, the applicant currently implements QMS, Revision 5, which commits to NQA-1-1994. In October 2008, the NRC conducted a QA implementation inspection to verify that design activities conducted for the AP1000 project comply with Revision 5 to the QMS and with the requirements of Appendix B to 10 CFR Part 50 and 10 CFR Part 21, "Reporting of defects and noncompliance." The inspectors verified that Revision 5 of the QMS is adequately implemented and identified three nonconformances to the requirements of Appendix B to 10 CFR Part 50. The applicant has responded to the nonconformances and addressed the actions to correct and prevent recurrence of these nonconformances in a letter dated February 3, 2009. The NRC determined that the actions to correct and prevent recurrence of these nonconformances were adequately addressed by the applicant, as described in a letter dated February 20, 2009, describing the NRC's review of the corrective actions.

The NRC staff identified Open Item OI-SRP 17.3-CQVP-01 for additional inspections of the applicant's implementation of the QMS as it relates to the AP1000 project, if necessary. The NRC has determined that additional inspections are not required to support the licensing decision of the AP1000 DCD Revision 17. Open Item OI-SRP 17.3-CQVP-01 is, therefore, closed.

As required by 10 CFR 52.63, "Finality of standard design certifications," the applicant provided justification for TR-109 in its letter dated May 26, 2007. The applicant stated that DCD Revision 16 includes changes to the certified design information that have resulted from inputs provided by the combined license (COL) applicants that will reference the AP1000 DCD. The applicant stated that the consensus group of AP1000 COL applicants has confirmed all of these changes, and the changes will result in an increased standardization of the certification information that all AP1000 COL applicants will adopt. Based on the NRC staff review of QMS, Revision 5, the NRC staff concludes that the adoption of NQA-1-1994 will result in increased standardization of the design certification (DC) information.

17.3.3 Conclusion

Based on the NRC staff's previous approval and subsequent inspection of the implementation of Revision 5 of the QMS, the NRC staff concludes that the applicant's adoption of NQA-1-1994 is acceptable for Revision 17 of the AP1000 DCD. Based on its review, the NRC staff finds that the QMS, as described in the AP1000 DCD Revision 17, meets the criteria of 10 CFR 52.63(a)(1)(vii) and Appendix B to 10 CFR Part 50 and is, therefore, acceptable.

17.4 Reliability Assurance Program During the Design Phase

17.4.1 Introduction

In the amendment to the AP1000 DCD, the applicant proposed changes to the description of the AP1000 reliability assurance program (RAP). The program has two parts. The first, accomplished during the design phase, is called the design reliability assurance program (D-RAP). The second, formerly called the operational reliability assurance program (O-RAP), is no longer required. References to the O-RAP were removed and operational phase reliability assurance activities (OPRAAs) that accomplish the objectives of the RAP after initial fuel load are now described.

The DCD amendment also clarified the association between the maintenance rule template and Nuclear Energy Institute (NEI) document NEI 07-02A, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52.” The DCA added certain structures, systems, and components (SSCs) to the D-RAP and updated the basis for the SSCs inclusion in the D-RAP as a result of updates to the risk analysis and decisions of the applicant’s expert panel.

Finally, the DCD amendment proposed a change to the inspection, test, analysis and acceptance criteria (ITAAC). The staff documented its evaluation of this change in Section 17.6, “Tier 1 Information.”

In addition to reviewing the amended DCD, the staff reviewed AP1000 combined license standard technical reports TR-117, APP-GW-GLR-117, Revision 1, “Incorporation of the Maintenance Rule,” and TR-132, APP-GW-GLN-132, “Changes to D-RAP Component List.” The staff also reviewed applicable sections of TR-134. These sections identify changes to Chapter 17 of the AP1000 DCD. Related changes also appear in Chapter 14 and Chapter 16 of DCD Tier 2, as well as in Section 3.7 of DCD Tier 1. This information is generic to the design and is expected to apply to all COL applications that reference the AP1000 DC.

17.4.2 Evaluation

The RAP has two stages. The first is referred to as the D-RAP. The second stage applies to reliability assurance activities for an operating plant; the objective during this stage is to ensure that the reliability of SSCs within the scope of the RAP is maintained during plant operations. Programs that are required by other regulations accomplish these reliability assurance activities. A separate, duplicative program would not be beneficial. Accordingly, Item E of the staff requirements memorandum (SRM) dated June 28, 1995, associated with SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084),” dated May 22, 1995, states that no separate O-RAP is required for licensees under 10 CFR Part 52, “Licenses, certifications, and approvals for nuclear power plants.”

Instead, the applicants are to incorporate the activities of the RAP after the design phase into existing plant programs. The following are examples of programs that include OPRAAs:

- maintenance rule
- QA
- inservice testing
- inservice inspection
- technical specifications surveillance requirements
- AP1000 investment protection short-term availability controls
- site maintenance

The OPRAA assignment (of activities formerly identified as within the O-RAP) is consistent with the guidance given in SECY-95-132 and with Section 17.4 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants.” For this reason, the staff finds the OPRAA assignments to be acceptable. This change also meets the requirement of 10 CFR 52.63(a)(1)(vii) in that it “[c]ontributes to increased standardization of the certification information.”

The applicant has deleted from the DCD Section 17.4.7.2, “D-RAP, Phase II”; Section 17.4.7.2.1, “Information Available to Combined License Applicant”; and Section 17.4.7.3, “D-RAP, Phase III.” The applicant justified these deletions by the incorporation of the generic final safety analysis report (FSAR) template in NEI 07-02A. The staff has endorsed NEI 07-02A and determined that it provides an acceptable method for complying with the requirement in 10 CFR 52.79(a)(15), “Contents of applications; technical information in final safety analysis report.” Specifically, FSARs must describe the program for monitoring the effectiveness of maintenance to meet the requirements of 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants” (also known as the Maintenance Rule), as well as how the program is to be implemented. This will satisfy the acceptance criteria of NUREG-0800 Section 17.6, “Maintenance Rule,” and is, therefore, acceptable to the staff. This change also meets the requirement of 10 CFR 52.63(a)(1)(vii) in that it “[c]ontributes to increased standardization of the certification information.”

The applicant has deleted four sections of the AP1000 DCD that addressed COL Information Items 17.5.3, 17.5.5, 17.5.6, and 17.5.7.

COL Information Item 17.5.3 states that “The COL applicant or holder will establish [probabilistic risk assessment] PRA importance measures, the expert panel process, and other deterministic methods to determine the site-specific list of SSCs under the scope of the RAP.” NEI 07-02A, Section 17.X.1.1, “Maintenance rule scoping per 10 CFR 50.65(b),” addresses this information item. Section 17.X.1.1 provides guidance on PRA insights for the SSCs, the establishment of the expert panel, the process for updating and maintaining the Maintenance Rule scope and SSC classifications, and the use of other deterministic methods for scoping SSCs.

The AP1000 DCD, COL Information Item 17.5.5 states the following:

> The following activities are represented in Figure 17.4-1 as “Plant Maintenance Program.” The Combined License applicant is responsible for performing the tasks necessary to maintain the reliability of risk-significant SSCs. Reference 8 [NUREG/CR-5695, “A Process for Risk-Focused Maintenance,"] contains examples of cost-effective maintenance enhancements, such as condition monitoring and shifting time-directed maintenance to condition-directed maintenance.

NEI 07-02A, Section 17.X.3, “Maintenance Rule Program Relationship with Reliability Assurance Activities,” addresses this information. Section 17.X.3 provides guidance on the implementation of operational programs for reliability assurance. These programs include QA, Maintenance Rule, maintenance, surveillance testing, inservice inspection, and inservice testing.

COL Information Item 17.5.6 states, “The Maintenance Rule (10 CFR 50.65) is relevant to the Combined License applicant’s maintenance activities in that it prescribes SSC performance-related goals during plant operation.”

NEI 07-02A addresses performance-related goals throughout the document. Specifically, Section 17.X.1.1 provides guidance on how SSCs in the Maintenance Rule program are evaluated against performance criteria to determine if goals must be established. In Section 17.X.1.2, the guidance states that goals are established for SSCs classified as (a)(1) status. Section 17.X.1.4 specifies a periodic evaluation of the performance, condition monitoring, goals, and preventive maintenance activities for SSCs in the Maintenance Rule program.

The AP1000 DCD, COL information item 17.5.7 states the following:

> In addition to performing the specific tasks necessary to maintain SSC reliability at its required level, the O-RAP activities include:
>
> - Reliability data base—Historical data available on equipment performance. The compilation and reduction of this data provides the plant with source of component reliability information.
> - Surveillance and testing—In addition to maintaining the performance of the components necessary for plant operation, surveillance and testing provides a high degree of reliability for the safety-related SSCs.
> - Maintenance plan—This plan describes the nature and frequency of maintenance activities to be performed on plant equipment. The plan includes the selected SSCs identified in the D-RAP.

These bulleted items are covered under NEI 07-02A, Section 17.X.3. The first bulleted item is also captured in Section 17.X.4, "Maintenance Rule Program Relationship with Industry Operating Experience Activities."

The staff considers NEI 07-02A to be an acceptable way to address these COL information items. These information items are considered to have been addressed for COL holders that implement programs consistent with this guidance. This change also meets the requirement of 10 CFR 52.63(a)(1)(vii) in that it "[c]ontributes to increased standardization of the certification information."

Most components included in the D-RAP (listed in Table 17.4-1, "Risk-significant SSCs within the Scope of D-RAP" and in Tier 1, Table 3.7-1, "Risk-Significant Components") are listed on the basis of a computed measure of risk importance. Specifically, a basic event with a risk achievement worth (RAW) greater than 2.0 or a risk reduction worth (RRW) greater than 0.005 is considered important to risk, as are the associated SSCs. SSCs may be assigned to the D-RAP for other reasons (for example, an appropriately constituted expert panel judges that they are important). The D-RAP list identifies only the expert panel as the basis for the inclusion of certain components, which implies that these components do not meet the quantitative risk importance criteria. The staff requested additional information to justify this conclusion in four cases:

(1) Reactor coolant pump switchgear circuit breakers:

ECS ES 31	ECS ES 41	ECS ES 51	ECS ES 61
ECS ES 32	ECS ES 42	ECS ES 52	ECS ES 62

In a letter dated September 5, 2008, the applicant provided additional information to justify the classification of the reactor coolant pump (RCP) without reference to the

RAW. The failures of the RCP switchgear circuit breakers are captured in the following events.

RC1CB051GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB052GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB053GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB054GO	PUMP A FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB061GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB062GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB063GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN
RC1CB064GO	PUMP B FAILS TO TRIP—BREAKER FAILS TO OPEN

In addition, the applicant stated the following:

> Fault tree logic requires all four pumps to shutdown; however, there are two breakers for each pump with only one of two breakers necessary to shutdown one pump. Consequently, CCF of the RCP switchgear circuit breakers is not modeled and, therefore, the RCP circuit breakers are in the D-RAP (Table 17.4-1) only due to the recommendation of the expert panel (EP).

The DCD for the certified design reported that following an event such as a loss-of-coolant accident (LOCA), steam generator tube rupture, or a stuck-open valve in the main steam line, RCPs must trip, or operator action will be required to achieve safe shutdown. Failure to trip RCPs can prevent operation of the core makeup tanks. Consequently, the staff expects a common-cause failure (CCF) to trip all RCPs to be significant, even if it does not lead to core damage in large LOCAs and some medium LOCAs.

This was confirmed in APP-GW-GL-022, “AP1000 Probabilistic Risk Assessment,” Revision 8 (the PRA report), which reported the CCF of RCP circuit breakers to open, represented by event RPX-CB-GO (see Table 29-2, “Common-Cause Failure Calculations,” page 29-22). Item 36 in Table 50-23 of the same report, “Risk Importances Sorted by Risk Increase,” shows RPX-CB-GO to be a member of more than 400 cutsets with a RAW greater than 50. It appears in reactor coolant loop, RCN, and RCT fault trees. The NRC staff identified this as Open Item OI-SRP17.4-SPLA-01.

In a letter dated May 13, 2009, the applicant proposed a revision to Table 17.4-1, indicating that these breakers are identified as risk-significant because of high RAW for CCF. The staff finds this to be acceptable and considers the open item to be resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

(2) 125-volt direct current 24-hour batteries, inverters, and chargers

The batteries provide power for the protection and safety monitoring system and safety-related valves. In a letter dated September 5, 2008, the applicant reported that the risk-significance criteria are not met by the basic events for these components without considering common cause. The risk-significance criteria for including the inverters, batteries, and battery chargers based on the PRA modeling of common cause for these components support inclusion in the D-RAP. The rationale for inclusion is changed to RAW/CCF for 125-volt dc 24-hour batteries, inverters, and chargers.

A subsequent design modification changed the voltage of the 1E direct current (dc) system to 250 volts direct current (Vdc). The staff's evaluation of this modification is documented in Section 8.3.2 of this safety evaluation. This had no impact on the way the system is modeled in the PRA and so PRA insights and results were not affected. In a letter dated April 20, 2010, the applicant proposed to revise Table 17.4-1 in Tier 2, reflecting the changes to the design and adding the associated 1E 250 Vdc buses to the explicit scope. This makes the table consistent with the amended design and it is, therefore, acceptable to the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

(3) In-containment refueling water storage tank vents

These vents (PXS-MT-03) provide a pathway to vent steam from the in-containment refueling water storage tank (IRWST) into the containment. In a letter dated September 5, 2008, the applicant reported that the method of modeling documented in APP-GW-GL-022 had been revised in the change discussed in TR-102, APP-GW-GLR-102, "AP1000 Probabilistic Risk Assessment Update Report."

The risk significance of IRWST vents based on the new PRA modeling for large-release frequency (LRF) supports inclusion in the D-RAP. The rationale for inclusion now reflects RAW for the IRWST vents. This is acceptable to the staff.

(4) Motor-operated valves in automatic depressurization system stages 1, 2, and 3

In the AP1000 DC PRA, the applicant reported a RAW value of greater than 40 for the event that represents failure-to-open for 32 combinations of three motor-operated valves (MOVs) in automatic depressurization system (ADS) stages 2 and 3.

In a letter dated September 5, 2008, the applicant reported that the criteria for including MOVs in ADS stages 1, 2, and 3 based on the PRA modeling (risk significance) of basic events before consideration of common cause for these components does not support inclusion of these MOVs in the D-RAP.

However, two common-cause groupings model CCFs of the ADS MOVs. One models the CCF to operate the MOVs of the ADS first, second, and third stages. The criteria for including MOVs in ADS stages 1, 2, and 3 based on LRF risk significance of this CCF supports their inclusion in the D-RAP. The other grouping models the CCFs of 32 combinations of three MOVs in stages 2 and 3 to fail to operate. The criteria for including MOVs in ADS stage 2 and 3 based on CDF and LRF risk significance supports inclusion of stage 2 and stage 3 MOVs in the D-RAP.

The rationale for including these MOVs in D-RAP now reflects RAW/CCF for MOVs in ADS stages 1, 2, and 3. This is acceptable to the staff.

Several additions to the D-RAP have been identified on the basis of risk significance in the internal events Level 1 PRA:

- chemical and volume control (CVS) letdown discharge isolation valve inside reactor containment (CVS-PL-V045)
- CVS control letdown discharge isolation valve outside reactor containment (CVS-PL-V047)
- a new diverse actuation system cabinet outside the main control room
- 6,900-volt alternating current buses ECS-ES-1 and ECS-ES-2, which are power buses fed by the onsite diesel generators and offsite power
- normal residual heat removal system (RNS) stop check valves (RNS-PL-V007A/B and RNS-PL-V015A/B) on the discharge of the RNS pumps, which prevent backflow from the reactor coolant system (RCS)

In addition, two components have been added on the basis of the Level 2 PRA:

- RNS check valve (RNS-PL-V013), which provides a flow path from the RNS pumps to the RCS
- RNS check valve (RNS-PL-V056), which provides a flow path from the cask loading pit to the RNS pump inlet

These additions are consistent with DCD Section 17.4.7.1.5, “SSCs to be Included in D-RAP,” and thus they are acceptable to the staff.

17.4.3 Conclusion

The applicant proposes changes to the RAP, described in the AP1000 DCA request. Both the removal of references to the O-RAP and the description of OPRAAs that complement the D-RAP to accomplish RAP objectives are consistent with NUREG-0800 Section 17.4. For this reason, they are acceptable to the staff.

The staff has endorsed adoption of NEI 07-02A, which allows closure of certain COL information items. The staff considers COL Information Items 17.5.3, 17.5.5, 17.5.6, and 17.5.7 to be closed for COL holders that implement programs consistent with this guidance.

The applicant added certain SSCs to the D-RAP and updated the basis for SSC inclusion in the D-RAP to include updates to the risk analysis and decisions of the applicant’s expert panel. These changes result from proposed design changes and updated risk analysis. They are consistent with the methodology previously approved and, therefore, acceptable to the staff.

These changes meet the requirement of 10 CFR 52.63(a)(1)(vii) in that each one “[c]ontributes to increased standardization of the certification information.”

17.6 Tier 1 Information

17.6.1 Information

Certain SSCs were added to the AP1000 D-RAP. The basis for SSC inclusion in the D-RAP was revised in some cases as a result of updates to the risk analysis and decisions of the applicant's expert panel.

In addition, the applicant proposed a change to the ITAAC for D-RAP.

In addition to reviewing the amended DCD, the staff reviewed TR-132. The staff also reviewed applicable sections of TR-134 and APP-GW-GLE-007, "ITAAC Changes" (Impact Document 07). In these, changes to Section 3.7 of Tier 1 of the AP1000 DCD are identified. Related changes also appear in Chapters 14, 16, and 17 of DCD Tier 2. This information is generic to the design and is expected to apply to all COL applications that reference the AP1000 DC.

17.6.2 Evaluation

Several additions to the D-RAP have been identified on the basis of risk significance in the internal events Level 1 PRA:

- CVS letdown discharge isolation valve inside reactor containment (CVS-PL-V045)
- CVS letdown discharge isolation valve outside reactor containment (CVS-PL-V047)
- Normal RNS stop check valves (RNS-PL-V007A/B and RNS-PL-V015A/B) on the discharge of the RNS pumps, which prevent backflow from the RCS

In addition, two components have been added on the basis of the Level 2 PRA:

- RNS check valve (RNS-PL-V013), which provides a flow path from the RNS pumps to the RCS.
- RNS check valve (RNS-PL-V056), which provides a flow path from the cask loading pit to the RNS pump inlet.

The nominal voltage of the 24-hour batteries was changed from 125 Vdc to 250 Vdc. (The staff's review of this modification is documented in Section 8.3.2 of this safety evaluation.)

Finally, the applicant revised the method by which certain SSCs were identified to clarify the rationale for their inclusion.

Subject to confirmation, the additions and changes will be consistent with DCD Section 17.4.7.1.5, "SSCs to be Included in D-RAP." They will be reflected in DCD Tier 1, Section 3.7, "Design Reliability Assurance Program," Table 3.7-1, "Risk-Significant Components," and Table 2.2.1-1 (untitled). The staff finds the proposed changes to Tier 1 will make Tier 1 and Tier 2 consistent and, therefore, they are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In APP-GW-GLE-007, "ITAAC Changes," (Impact Document 07), the applicant proposed an alternative to the existing D-RAP ITAAC. In Table 3.7-3, “Inspections, Tests, Analyses and Acceptance Criteria,” the approved existing acceptance criterion is as follows:

> A report exists and concludes that the estimated reliability of each as-built component identified in Table 3.7-1 is at least equal to the assumed reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.

To this, the applicant proposed to add:

> For an as-built component with reliability less than the assumed reliability an evaluation shall show that the net effect of as-built component reliabilities does not reduce the overall reliability. Or, an evaluation shall show that there is not a significant adverse effect on the core melt frequency or the large release frequency in the PRA applicable to the plant.

The staff could not conclude that the proposed alternative would satisfy the expectations documented in the SRM dated June 28, 1995, for SECY-95-132.

D-RAP ITAAC should provide assurance that the reliability and availability of risk-significant SSCs are consistent with the certified design (subject to deviations and plant-specifics approved in the COL and reflected in the FSAR). In RAI-SRP17.4-SPLA-04, the staff requested that the applicant propose an alternative D-RAP ITAAC that provides reasonable assurance that the plant is designed and will be constructed in a manner that is consistent with the key assumptions and risk insights for risk-significant SSCs within the scope of D-RAP. The NRC staff identified this as Open Item OI-SRP17.4-SPLA-04.

In a letter dated March 30, 2010, the applicant proposed a revised D-RAP ITAAC. The applicant proposed new language in Table 3.7-3, “Inspections, Tests, Analyses and Acceptance Criteria.” The proposed design commitment now reads as follows: “The D-RAP ensures that the design of SSCs within the scope of the reliability assurance program (Table 3.7-1) is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability).” The ITAAC is to be completed by an analysis: “An analysis will confirm that the design of RAP SSCs identified in Table 3.7-1 has been completed in accordance with applicable D-RAP activities. The following acceptance criteria are provided:

> An analysis report documents that safety-related SSCs identified in Table 3.7-1 have been designed in accordance with a 10 CFR 50 Appendix B quality program.
>
> An analysis report documents that nonsafety-related SSCs identified in Table 3.7-1 have been designed in accordance with a program which satisfies quality assurance requirements for SSCs important to investment protection.

The staff finds that satisfying this commitment will provide adequate assurance that the design products used to procure, manufacture, transport, store, install, inspect, and test risk-significant SSCs are traceable to the licensed design. It is consistent with SRM for SECY-95-132. This component of ITAAC complements the others in providing confidence that the as-built plant is, at the time of initial fuel loading, consistent with the certified design. For this reason, it is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

17.6.3 Conclusion

Revisions to DCD Tier 1 Table 3.7-1, "Risk-Significant Components," and Table 2.2.1-1 (untitled) have been proposed. With these changes, the Tier 1 information corresponds to the Tier 2 information in Table 17.4-1, "Risk-Significant SSCs within the Scope of D-RAP," as it has been clarified and updated to reflect the amended design. These additions and changes are consistent with the previously approved methodology and, therefore, they are acceptable to the staff.

The staff reviewed the applicant's proposed change to D-RAP ITAAC. The staff finds that the proposed D-RAP ITAAC will provide assurance that the reliability and availability of risk-significant SSCs are consistent with the certified design (subject to deviations and plant specifics approved in the COL and reflected in the FSAR). It is consistent with the SRM for SECY-95-132 and is, therefore, acceptable to the staff.

These changes meet the requirement of 10 CFR 52.63(a)(1)(vii) in that each one "[c]ontributes to increased standardization of the certification information."

18. HUMAN FACTORS ENGINEERING

Westinghouse has submitted information in support of its design certification (DC) amendment application that Westinghouse considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5). Westinghouse has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears within "square brackets" as follows:

[ ]

The complete text of this chapter, containing proprietary information can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

18.2 Element 1: Human Factors Engineering Program Management

18.2.5 AP1000 Human Factors Engineering Program Plan (no comparable NUREG-1793 section)

18.2.5.1 Summary of Technical Information

The applicant added APP-OCS-GBH-001, Revision 0, "AP1000 Human Factors Engineering Program Plan," issued January 2008 (hereafter referred to as the AP1000 Human Factors Engineering (HFE) Program Plan, as a reference in Section 18.2.6.1 of the design control document (DCD) to establish completion of combined license (COL) Information Item 2-1. In technical report (TR)-90 (APP-GW-GLR-090, Revision 0, "Strategy for the Closure of the AP1000 Design Control Document, Chapter 18, 'Human Factors Engineering (HFE) COL Information Items,'" issued February 2007), the applicant stated that the AP1000 HFE Program Plan captures the technical content discussed in DCD Section 18.2, and describes implementation methods for incorporating HFE into the AP1000 design process. It is designed to serve as an implementation manual for the engineering staff.

18.2.5.2 Staff Evaluation

The staff determined that this additional reference is consistent with Revision 15 of the AP1000 DCD. It describes the scope of the HFE program in terms of each of the elements identified in NUREG-0711, "Human Factors Engineering Program Review Model," issued February 2004. The reference also describes the processes used to incorporate HFE into the AP1000 design process.

18.2.5.3 Conclusion

The staff concludes that this change does not affect the conclusion in Section 18.2.4 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” issued September 2004).

The AP1000 HFE Program Plan continues to implement NUREG-0711 guidelines.

18.2.6 Evaluation of COL Information Item 18.2-1 (no comparable NUREG-1793 section)

18.2.6.1 Summary of Technical Information

COL Information Item 18.2-1 states the following:

> The COL applicant referencing the AP1000 certified design is responsible for the execution of a U.S. Nuclear Regulatory Commission (NRC)-approved HFE Program.

In DCD Revision 17, the applicant stated:

> [The] AP1000 Human Factors Engineering Program Plan fully captures the information certified in Section 18.2 and provides execution guidance for the NRC-approved HFE Program. The ongoing confirmation that the AP1000 HFE Program Plan is being executed as required is demonstrated by fulfillment of the other COL Information Items in Chapter 18. The final confirmation that the HFE Program Plan has been executed will be demonstrated by completion of the ITAAC [Inspections, Tests, Analyses and Acceptance Criteria] (Tier 1 Material, Table 3.2-1, Items 1 to 13).

18.2.6.2 Evaluation

From a program overview perspective, the applicant used Revision 17 to document changes in the status of a number of COL information items and inspection, test, analyses, and acceptance criteria (ITAAC). TRs provided for staff review include the supporting documentation. When the TRs indicate that partial progress has been made and additional work to address information items is ongoing, the staff evaluated redundancy between the COL information item and the ITAAC. If the staff identified redundancy, then the COL action item was closed. In all cases, the review ensured that final design product completion was appropriately tracked. The staff identified all documents used to conclude that the NUREG-0711 criteria were satisfactorily implemented, and Westinghouse docketed the documents.

With respect to COL Information Item 18.2-1, the staff determined that the information item is closed based on the following:

(1) The AP1000 HFE Program Plan is consistent with AP1000 DCD, Revision 15. It describes the scope of the HFE program in terms of each of the NUREG-0711 elements. It provides additional information on where and how the overall design process should use HFE guidance and, thus, provides reasonable assurance that the applicant will implement and undertake the required HFE activities for the AP1000 design at the most appropriate time in the project schedule. This program element requires no additional product development.

(2) COL information items in other sections and the ITAAC listed in Table 3.2-1 address specific HFE design products that require completion. Retaining this generic information item is redundant with the remaining open information items and ITAAC.

(3) The applicant must implement the HFE verification and validation (V&V) program in accordance with ITAAC Table 3.2-1, Item 1 (DCD Revision 17). This includes five specific tasks that validate and verify HFE program implementation and concludes with an as-built inspection of the human-system interfaces (HSIs) as constructed at the time of plant startup. The combination of these actions provides better verification of field implementation than would be accomplished under this COL information item.

18.2.6.3 Conclusion

The staff concludes that COL Information Item 18.2-1 is redundant to existing ITAAC included in Tier 1, Table 3.2-1. Consequently this COL information item is closed. ITAAC Item 1 (DCD Revision 17) will verify the execution of the NRC-approved HFE program.

18.2.7 Evaluation of COL Information Item 18.2-2 (no comparable NUREG-1793 section)

18.2.7.1 Summary of Technical Information

COL Information Item 18.2-2 states the following:

> Specific information regarding the location of the emergency operations facility and emergency operations facility communications will be provided by the Combined Operating License applicant to address the Combined License information requested in this subsection.

18.2.7.2 Evaluation

The applicant stated in TR-134 (APP-GW-GLR-134, Revision 4, “AP1000 Standard COLA Technical Report,” issued March 2008) that TR-136 (APP-GW-GLR-136, Revision 1, “AP1000 Human Factors Program Implementation for the Emergency Operations Facility and Technical Support Center,” issued October 2007) partially addresses the information requested by this information item. In TR-136, the applicant described the method used to apply the AP1000 HFE Program Plan to technical support centers (TSCs) and emergency operations facilities (EOFs) used to support AP1000 plants and stated that the COL applicant has overall responsibility for the human factors adequacy of the TSC and EOF. In APP-OCS-GGR-110-P, Revision 1, “AP1000 Technical Support Center and Emergency Operations Facility Workshop,” issued February 2008, the applicant described in detail how it developed the information in TR-136.

In TR-136 and subsequently in AP1000 DCD Revision 17, the applicant made changes to this COL information item that deleted HFE design responsibilities that were included in the previously approved COL information item in the DCD, Revision 15. In response to request for additional information (RAI)-SRP18-COLP-21, dated July 31, 2009, the applicant removed EOF and TSC location requirements and added responsibilities for EOF and TSC human factors attributes.

Deletion of location requirements is acceptable because the HFE design is not dependent on location. The location of the EOF and TSC is subject to regulatory guidance. This is addressed in Section 13.3, “Emergency Planning,” of this report.

The addition of COL responsibility for defining EOF and TSC human factors attributes is consistent with the intent of the original, approved COL information item and ensures that the HFE design outside the scope of the AP1000 DCD is addressed. In a subsequent revision to the AP1000 DCD, the applicant incorporated the RAI response into the DCD text, which resolves this issue.

From the program description provided in TR-136 and APP-OCS-GGR-110-P, the NRC staff noted a well-structured and disciplined assessment of the HFE requirements applicable to the TSC and EOF. The following examples demonstrate how the applicant used the AP1000 HFE Program Plan and appropriate regulations to identify the HFE design requirements of the TSC/EOF:

- Westinghouse and utility personnel worked together to identify the functional requirements for the TSC/EOF. The diverse experience in this group supported a thorough evaluation.

- Westinghouse extracted specific requirements from the AP1000 DCD; the AP1000 HFE Program Plan; NUREG -0711, Revision 2; NUREG-0696, “Functional Criteria for Emergency Response Facilities,” issued February 1981; and NUREG-0654/FEMA-REP-1, Revision 1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” issued March 2002. These documents serve as the basis for identifying the TSC/EOF functional requirements. Identification of functional requirements is one of the basic steps required in the AP1000 HFE Program Plan and NUREG-0711. APP-OCS-GGR-110-P provides complete documentation of how Westinghouse identified applicable functions.

- Westinghouse and utility representatives conducted an operating experience review (OER). Application of lessons learned from operating experience is one of the basic steps required in the AP1000 HFE Program Plan and NUREG-0711.

- Westinghouse completed a task analysis incorporating OER results, observations from emergency plan drills at V.C. Summer and Harris nuclear sites, input from emergency procedures from four different utilities, and review comments from both Westinghouse and utility personnel. In TR-136, the applicant stated that it will capture the requirement for this task analysis in the Operational Sequence Analysis-2 (OSA-2) Implementation Plan. A task analysis is one of the basic steps required in the AP1000 HFE Program Plan, NUREG-0711, and OSA-2 incorporates accepted methods for performing task analyses.

- In accordance with TR-136, Section 2.4.4, Westinghouse has identified applicable HSI design guidelines from the AP1000 HSI design guidelines (APP-OCS-J1-002, Revision 0, “AP1000 Human System Interface Design Guidelines”) to promote the human factors design adequacy of the TSC/EOF design. This ensures that standard HSI design requirements will be applied to the appropriate elements of HSI design.

- EOF/TSC HFE design elements outside the AP1000 scope are addressed via a COL information item. This provides reasonable assurance that a complete HFE design will be achieved for these emergency facilities.

Based on the activities outlined above, the applicant's use of a tailored approach in applying the AP1000 HFE program to the TSC and EOF is solidly based on NUREG-0711. The applicant has documented the TSC and EOF task analysis results in APP-OCS-JOA-001, "HFE Analysis to Support TSC and EOF Design." Open Item OI-SRP18-COLP-18 tracked completion of the task analysis and documentation of the results. This open item has been addressed by issuance of the report, which satisfactorily summarizes task analysis results associated with the EOF and TSC and is closed.

18.2.7.3 Conclusion

The applicant has developed a sufficient basis for applying a tailored HFE program to the TSC and EOF and has documented the TSC and EOF task analysis results in the APP-OCS-JOA-001 report. The revised COL Information Item 18.2-2 accurately communicates the COL applicant's responsibility for HFE design of the EOF and TSC. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which incorporates the revised COL information.

18.2.8 Evaluation of Tier 1 Information—Design Commitment 3, ITAAC Table 3.2-1 (DCD Revision 15)

18.2.8.1 Summary of Technical Information

ITAAC Design Commitment 3 (DCD Revision 15) reads as follows:

> Design Commitment: The HSI design is performed for the operation and control system in accordance with the HSI design implementation plan.
>
> Inspection, Tests, and Analyses: An evaluation of the implementation of the HSI design.
>
> Acceptance Criteria: A report exists and concludes that the HSI design for the operation and control system was conducted in conformance with the implementation plan and includes the following documents:
>
> – Operation and Control Centers System Specification Document
>
> – Functional requirements and design basis documents for the alarm system, plant information system, wall panel information system, controls (soft and dedicated), and the qualified data processing subsystems
>
> – Design guideline documents (based on accepted HFE guidelines, standards, and principles) for the alarm system, displays, controls, and Anthropometrics
>
> – Design specifications for the alarm system, plant information system, wall panel information system, controls (soft and dedicated), and the qualified data processing subsystems

- Engineering test report document summarizing outcomes of each man-in-the-loop engineering test iteration performed to support HSI design

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.2.8.2 Evaluation

The staff did not find a one-to-one correlation between the list of completed documents in TR-82 (APP-GW-GLR-082, Revision 0, “Execution and Documentation of the Human System Interface Design Implementation Plan,” issued May 2007) and the AP1000 DCD, Tier 1, ITAAC Table 3.2, Design Commitment 3 (DCD Revision 15), acceptance criteria. Design documents were not identified for the following areas:

- functional requirements and design-basis documents for the plant information system
- functional requirements and design-basis documents for controls (soft and dedicated)
- functional requirements and design-basis documents for the qualified data processing subsystems

RAI-SRP18-COLP-05 requested clarification of the discrepancy. The applicant’s response, dated May 28, 2008, indicated that terminology changes resulted in the inclusion of the areas listed above in the “Distributed Control and Information System” (APP-OCS-J1-010, “AP1000 Display System Functional Requirements”). The staff found that this document includes the functional requirements and design-basis information for the systems listed above. The staff concluded that this change was limited to renaming and reorganizing information to improve clarity and did not affect the intent of the ITAAC.

Open Item OI-SRP18-COLP-01A, identified that the applicant had not completed all the design specifications listed in the ITAAC. These design specifications were subsequently completed along with specifications for systems not listed.

The staff reviewed the completed documents referenced in TR-82, along with the information provided in the RAI, and concluded that these documents appropriately implement the HSI design implementation plan, as described in AP1000 DCD, Revision 17. This included the documents referenced in the ITAAC. Clarity was consistently good across the procedural hierarchy, and the specificity of design requirements had increased in the transition from the functional design level to design specifications. These procedures provide reasonable assurance that the design process will effectively implement standardized HFE design requirements. Based on these results, OI-SRP18-COLP-01A has been satisfactorily addressed and is closed. The staff evaluates the translation of these design documents to a physical design as part of ITAAC Table 3.2-1 Design Commitment 1b (DCD Revision 17).

18.2.8.3 Conclusion

The staff concludes that the proposed change to delete Design Commitment 3 of the ITAAC is supported by the quality of the design documents that have been produced. The design documents provide the level of detail needed to provide reasonable assurance that the HFE design will be effectively implemented within the control room, remote shutdown station, and local control stations. Design Commitment 3 in ITAAC Table 3.2-1 (DCD Revision 15) is closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.5 Element 4: Task Analysis

18.5.5 Evaluation of Operational Sequence Analysis-2 Implementation Plan and Results Summary

18.5.5.1 Summary of Technical information

In AP1000 DCD, Revision 17, the discussion of OSA-2 deleted the description of a specific theoretical model for evaluating operator workload measures, but still committed to conducting an evaluation of the effect of the HSI design and the task demands on operator workload. In TR-81 (APP-GW-GLR-081, Revision 1, "Closure of COL Information Item 18.5-1, Task Analysis," issued May 2007) and in the RAI-TR81-COLP-01 response dated January 29, 2008, the applicant indicated that APP-OCS-J1R-210, Revision 1 "AP1000 Operational Sequence Analysis (OSA-2) Implementation Plan," would identify the most appropriate task analysis methods to use.

18.5.5.2 Evaluation

The staff reviewed the OSA-2 Implementation Plan, which describes the applicant's methods for analyzing the collected task sequence information needed to satisfy the four issues addressed in the DCD: (1) completeness of available information; (2) time to perform tasks; (3) operator workload analysis; and (4) operational crew staffing. The staff concludes that it is acceptable to remove the prescriptive language from the DCD because the applicant provided a robust implementation plan containing detailed information describing how to conduct an OSA analysis, the tasks that should be part of the analysis, and the expected results from the analysis.

The staff also reviewed APP-OCS-J1R-220, Revision B, "AP1000 Operational Sequence Analysis (OSA-2) Summary Report," which describes the results of conducting the activities described in the implementation plan.

18.5.5.3 Conclusion

Based on its review of the implementation plan and the summary report, the staff has determined that sufficient information exists to address the NUREG-0711 review criteria as described in the COL closure section below.

18.5.6 Evaluation of COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-3)

18.5.6.1 Summary of Technical Information

COL Information Item 18.5-1 states the following:

> Combined License applicants referencing the AP1000 certified design will address the execution and documentation of the task analysis implementation plan presented in Section 18.5.

Appendix F of NUREG-1793 broke this COL information item into two pieces and reworded the information item so that it reads:

> FSER Item 18.5.3-3: The staff reviewed the applicant's task analysis at an implementation plan level of detail; finished products to complete the element were not available for review, but the methodology for conducting a complete task analysis was evaluated. The COL applicant will use this methodology to conduct a complete HFE task analysis after design certification.
>
> FSER item 18.5.3-2: The COL applicant will utilize the information from the AP1000-specific task analysis in the development of its procedures and training programs.

This section addresses execution of the task analysis implementation plan, which Westinghouse completed. TR-81 was submitted to document the task analysis results. The report recommends a revision to Tier 1 of the DCD ITAAC to reflect completion of the AP1000 function-based task analysis and provides a basis for closure of COL Information Item 18.5-1. The applicant also provided the OSA-2 Implementation Plan, which describes the methodology used to conduct the second round of OSA.

18.5.6.2 Evaluation

The staff evaluated the information provided by the applicant in the OSA-2 Implementation Plan and the OSA-2 Summary Report against acceptance criteria from NUREG-0711, Revision 2.

NUREG-0711, Section 5.4(1), states Criterion 1 as the following:

> The scope of the task analysis should include the following:
>
> - selected representative and important tasks from the areas of operations, maintenance, test, inspection, and surveillance
> - a full range of plant operating modes, including startup, normal operations, abnormal and emergency operations, transient conditions, and low power and shutdown conditions
> - Human Actions (HAs) that have been found to affect plant risk by means of PRA importance and sensitivity analyses should also be considered risk-important

- where critical functions are automated, all human tasks, including monitoring of the automated system and execution of backup actions if the system fails

Evaluation of Criterion 1

The staff reviewed OSA-2 Summary Report, which provides the results of the OSA-2 for the AP1000 design in accordance with the OSA-2 Implementation Plan. The OSA-2 Implementation Plan summarizes [ ] components and the corresponding [ ] maintenance, test, inspection, and surveillance (MTIS) tasks used for the task analysis. The inclusion of representative and important tasks from the areas of operations, MTIS during OSA-2 implementation satisfies the requirements in the first bullet of NUREG-0711 Criterion 1.

The OSA-2 Implementation Plan identified [ ] risk-important tasks, including tasks during normal operations, emergency and abnormal operations, and shutdown conditions. The inclusion of these tasks within the scope of the task analysis implementation plan satisfies the requirements of the second bullet of NUREG-0711 Criterion 1.

As described in the AP1000 DCD, the applicant performed OSA-2 for a representative subset of tasks including risk-important human actions, risk-important tasks, and tasks that have human performance concerns. The applicant used human reliability analysis (HRA) to identify [ ] scenarios and associated tasks described in the implementation plan as risk-important tasks. This is an acceptable method for identifying risk-important tasks. The applicant used probabilistic risk assessment (PRA) to identify estimated timeframes for completing the tasks, as well as the beginning and ending steps. For example, the plan discusses tasks associated with [ ] an [ ], and a [ ]. These events are considered to be within the design basis, and risk-important tasks are associated with them. The identification and inclusion of these risk-important tasks within the scope of the OSA-2 Implementation Plan satisfies the third bullet of NUREG-0711 Criterion 1.

The OSA-2 Implementation Plan discusses [ ] tasks identified during OSA-1 as having human performance concerns. These [ ]. The selection of tasks that have associated human factors concerns demonstrates that the applicant has chosen to analyze critical automated functions. The implementation plan describes the backup actions to be taken in case of a failure. Analysis of automated system failures and backup actions during OSA-2 satisfies the fourth bullet of NUREG-0711 Criterion 1.

The staff has determined that the scope of the task analysis is consistent with NUREG-0711 Criterion 1.

NUREG-0711, Section 5.4 (2), states Criterion 2 as the following:

> Tasks should be linked using a technique such as operational sequence diagrams. Task analyses should begin on a gross level and involve the development of detailed narrative descriptions of what personnel have to do. The analyses should define the nature of the input, process, and output needed by and of personnel.

Evaluation of Criterion 2

Consistent with the NUREG-0711 criterion, the applicant used the [ ] methodology to conduct two rounds of analysis. The first analysis, OSA-1, identified risk-important tasks, [ ] for safe operation and safe shutdown for the AP1000 design. OSA-2 used the tasks identified from OSA-1 to estimate [ ].

The staff reviewed the OSA-1 report titled "AP1000 Operational Sequence Analysis (OSA) Summary Report," Revision 0, (APP-OCS-J1R-120) and the implementation plans for OSA-1 (APP-OCS-J1R-110) and OSA-2, (APP-OCS-J1R-210). These reports describe the applicant's methods for analyzing the collected sequence information needed to satisfy the four issues identified in the DCD:

(1) Completeness of information: Establish the necessary information for successful task performance. The results of this analysis feed into the interface design process to ensure necessary information is available to the operator performing the task activities.

(2) Time to perform tasks: Establish that the operators will be able to complete tasks within the time available. This information is based on assumptions about the time required to access displays, select and actuate controls, etc. The OSA-2 Summary Report discusses that the generally acceptable range of "good" or appropriate operator workload is between 50 and 80 percent. A workload greater than 80 percent indicates a potential overload, while a workload less than 50 percent indicates a potential underload.

(3) Operator Workload: Establish the impact of task requirements and the HSI design on operator workload.

(4) Operational crew staffing: Establish staffing requirements. The results of the operator workload assessment and the identification of time constraints are used to review the adequacy of staffing assumptions, HSI design, task allocation and work organization.

Since the OSA-1 analysis was more general than the OSA-2 analysis, and the information from OSA-1 was used as input into OSA-2, the applicant's task analysis is consistent with NUREG-0711 Criterion 2 that the analysis should begin on a gross level (OSA-1) and become more detailed as the analysis proceeds.

Because of the overlap between Criteria 2 and 3, the staff presents its evaluation of the applicant's task analysis with regard to development of detailed narrative descriptions under Criterion 3 below and addresses the input, process, and output needed by and from personnel.

Based on its evaluation, the staff concludes that the applicant has satisfactorily met NUREG-0711 Criterion 2.

NUREG-0711, Section 5.4(3), states Criterion 3 as the following:

> The task analysis should be iterative and become progressively more detailed over the design cycle. It should be detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment.

Evaluation of Criterion 3

Westinghouse conducted OSA-1 and OSA-2 and described these analyses in APP-OCS-J1R-120, Revision 0, and APP-OCS-J1R-220, Revision B, respectively. The staff evaluated these documents and found that OSA-1 focused on specifying the operational requirements for a set of selected tasks. OSA-1 also identified risk-important tasks and thoroughly described the [ ]. An [ ] was developed to show information such as [ ]. The staff determined that OSA-1 lays the foundation for and describes the tasks that were analyzed during OSA-2. These analyses meet the NUREG-0711 criterion that the task analysis should be iterative and become progressively more detailed over the design cycle, because the task analysis was repeated and OSA-2 is more detailed than OSA-1.

The staff reviewed the task analysis documents to determine whether the task analyses conducted were detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment. The staff's evaluation addressed the input, process, and output needed by and from personnel (part of Criterion 2 above). OSA-2 Implementation Plan, Section 2.2.1 indicates that once the sequences for analysis have been identified, information related to the tasks in each sequence must be selected, including the following:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

- [ ]
- [ ]
- [ ]

To illustrate this aspect of the task analysis data collection process, the applicant provided an example task analysis for [ ] in the OSA-2 Summary Report. For one risk-important task, "[ ]" the applicant identified (1) [ ].

Based on its review of the process discussed in the applicant's OSA-2 Implementation Plan and the example provided, the staff concludes that the task analyses conducted were detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment.

Based on its evaluation, the staff concludes that the applicant has satisfactorily met NUREG-0711 Criterion 3.

NUREG-0711, Section 5.4(4), states Criterion 4 as the following:

> The task analysis should address issues such as:
>
> - the number of crew members
> - crew member skills
> - allocation of monitoring and control tasks to the (a) formation of a meaningful job, and (b) management of crew member's physical and cognitive workload.

Evaluation of Criterion 4

Section 2.3.3 of the task analysis implementation plan describes operator workload analysis as an evaluation of the effect of the HSI design and the demands on operator workload. The methodology used for assessing [ ].

Section 2.3.4 of the task analysis implementation plan discusses the analysis of operational crew staffing. When the OSA indicates [ ]. The applicant uses the results from the OSA to [ ]. The applicant provided an example of the [

].

The staff has concluded that Westinghouse has conducted a thorough task analysis using both OSA-1 and OSA-2 and has described in detail the results from its analysis. The OSA-2 analysis was conducted in accordance with the implementation plan, which addresses issues such as the number of crew members, crew member skills, and allocation of monitoring and control tasks.

NUREG-0711, Section 5.4 (5), states Criterion 5 as the following:

> The task analysis results should be used to define a minimum inventory of alarms, displays, and controls necessary to perform crew tasks based on both task and instrumentation and control requirements.

Evaluation of Criterion 5

The OSA-2 Implementation Plan, Section 2.3.1, states that the [ ]. The HSI design process uses the [ ] of OSA-2. This approach is consistent with the information described in Revisions 15 through 17 of the DCD and satisfies NUREG-0711 Criterion 5.

NUREG-0711, Section 5.4 (6), states the following as Criterion 6:

> The task analysis should provide input to the design of HSIs, procedures, and personnel training programs.

Evaluation of Criterion 6

The OSA-1 analysis identified inputs to the HSI design including display requirements, display design constraints, performance time constraints, inventory (alarms, controls, parameters), and display organization and navigation constraints. OSA-2 is performed as part of the design development process to understand the estimated operator workload, performance time estimates, staffing issues, and error potential associated with each task. The staff concludes that, as with OSA-1, the results of OSA-2 provide input to the design of HSIs by providing a set of requirements and constraints on operator task performance.

In NUREG-1793, Section 18.5.3, the staff identified COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-2), which states, “The COL applicant will use the information from the AP1000-specific task analysis in the development of its procedures and training programs.” In response to RAI-SRP18.5-COLP-01, dated May 28, 2008, the applicant referred to Sections 5.6 and 5.7 of the AP1000 HFE Program Plan (APP-OCS-GBH-001), which describes two documents: APP-OCS-GER-031, “The Incorporation of Human Factors Engineering into the Development of the AP1000 Plant Procedures,” and APP-OCS-GER-041, “The Incorporation of Human Factors Engineering into the Development of the AP1000 Plant Training Program.” According to the applicant’s response to the RAI, the purpose of these documents is to capture the operator training and procedure information identified in the task analyses. These reports ensure that information related to training and procedures is identified, recorded, and communicated to those responsible for the development of the training programs. Open Item OI-SRP18-COLP-17 tracked completion of these documents. The staff has reviewed the completed documents and determined that information from OSA-1 and OSA-2 analyses useful to procedures and training program development has been identified, extracted, and compiled such that it can be used as direct input by procedure and training developers. Accordingly, the staff finds that Criterion 6 has been satisfactorily addressed and the open item is closed.

NUREG-0711, Section 5.4 (7), states Criterion 7 as the following:

> Considerations should be addressed for plant modifications that are likely to affect HAs previously identified as risk-important, cause existing HAs to become risk-important, or create new actions that are risk-important.

Evaluation of Criterion 7

The applicant is not required to address the impact of plant modifications on risk-important HAs because Revision 17 of the AP1000 DCD applies to new plant construction.

18.5.6.3 Conclusion

In its evaluation of Revision 15 of the AP1000 DCD, the staff reviewed the function-based task analysis and OSA-1 results and concluded that the applicant had developed an acceptable task analysis implementation plan to satisfy the NUREG-0711 criteria for task analyses. The COL applicant was expected to use this methodology to conduct a complete task analysis after DC (Reference COL Action Item 18.5.3-3). To close this action item, Revision 17 of the AP1000 DCD referenced additional task analysis documents, which describe an implementation plan for conducting a second operational sequence analysis (OSA-2) and provide a summary report of the OSA-2 results. The OSA-2 Implementation Plan and OSA-2 Summary Report focus on risk-important human actions, tasks with high human performance concerns, and on MTIS activities. Based on its evaluation of Revision 17 of the DCD and the referenced reports, the staff concludes that the applicant's task analysis conforms to all applicable criteria from NUREG-0711, Section 18.5.

18.5.7 Evaluation of COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-2)

18.5.7.1 Summary of Technical Information

COL Information Item 18.5-1 from the DCD does not correlate well with its counterpart, NUREG-1793 Item 18.5.3-2 which states:

> The COL applicant will utilize the information from the AP1000-specific task analysis in the development of its procedures and training programs.

The DCD Information Item 18.5-1 (see previous section) focuses on documentation of task analysis results and the staff identified information item addresses application of that information.

18.5.7.2 Evaluation

In response to RAI-SRP18.5-COLP-01, the applicant referred to Sections 5.6 and 5.7 of the AP1000 HFE Program Plan, which describes two documents (APP-OCS-GER-031 and APP-OCS-GER-041). These documents capture the operator training and procedure information identified in the task analyses. They provide an acceptable vehicle for communicating this information to those responsible for the development of the procedure and training programs. This directly addresses the DCD related action to document the task analysis results.

Using task analysis results to support procedure program development is satisfactorily addressed in the writer's guides, which are discussed in Section 18.9.5.2 of this report.

Using task results to support training program development is not directly addressed in the DCD. When the DCD Revision 15 safety evaluation report (SER) was prepared, no action was taken to include an additional COL information item to reflect the conclusions in the SER. Addition of the action to DCD Revision 18 is considered unnecessary. The bases material has been made readily available. COL applicants can use this information as appropriate as they develop systems approach to training (SAT) based training programs in accordance with industry and regulatory guidance.

18.5.7.3 Conclusion

This COL information item is closed because APP-OCS-GER-031 and APP-OCS-GER-041 adequately communicate the task analysis results applicable to procedure and training program development. As mentioned earlier, this work was being tracked by Open Item OI-SRP18-COLP-17, which has been closed.

18.5.8 Evaluation of COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1)

18.5.8.1 Summary of Technical Information

COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1) states the following:

> [A] COL applicant referencing the AP1000 certified design will document the scope and responsibilities of each Main Control Room position, considering the assumptions and results of the task analysis.

The applicant submitted TR-52 (APP-GW-GLR-010, Revision 2, "AP1000 Main Control Room Staff Roles and Responsibilities," issued June 2007) as a basis for closing COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1).

18.5.8.2 Evaluation

TR-52 states that the applicant has fully addressed the COL information item. Revision 18 of the DCD incorporates the applicable changes. As described in Section 4.5 of TR-52, the role of the shift technical advisor (STA) for the AP1000 design, including the role of assessing possible significant plant abnormalities observed during normal operations, is consistent with the typical responsibilities of the STA as listed in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980.

The staff issued RAI-TR52-COLP-12 asking the applicant to further clarify the duties and responsibilities in some key areas, including the reactor operator (RO) and STA roles in communication and coordination. In its response, dated November 16, 2007, the applicant clarified the roles and responsibilities of the RO and STA, describing the responsibilities for all main control room (MCR) ROs to communicate with the MCR supervisor and local equipment operators (EOs) to ensure coordination of local unit evolutions with plant operations. The RAI response also described the responsibilities of the MCR supervisor to maintain awareness of directions given to the EOs and to evaluate any abnormal conditions or operating concerns reported by either the ROs or the EOs. The OSA-2 Implementation Plan and the OSA-2 Summary Report also address MCR responsibilities. These responsibilities conform to the requirements in 10 CFR 50.54, "Conditions of licenses."

18.5.8.3 Conclusion

The staff finds that TR-52 adequately describes the MCR staff roles. The applicant's response to RAI-TR52-COLP-12 addresses each section of the RAI not addressed in TR-52; including specifying how each STA responsibility matches with the list of 12 responsibilities in Appendix C to NUREG-0737. These documents in combination provide sufficient information to close COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1).

18.5.9 Evaluation of Tier 1 Information—Design Commitment 2, ITAAC Table 3.2-1 (DCD Revision 15)

18.5.9.1 Summary of Technical Information

ITAAC Design Commitment 2 reads as follows:

> Design Commitment: The applicant performs a task analysis in accordance with the task analysis implementation plan.
>
> Inspection, Tests, and Analyses: An evaluation of the implementation of the task analysis will be performed.
>
> Acceptance Criteria: A report exists and concludes that function-based task analyses were conducted in conformance with the task analysis implementation plan and include the following functions:
>
> – Control reactivity
> – Control reactor coolant system (RCS) boron concentration
> – Control fuel and cladding temperature
> – Control RCS coolant temperature, pressure, and inventory
> – Provide RCS flow
> – Control main steam pressure
> – Control steam generator inventory
> – Control containment pressure and temperature
> – Provide control of main turbine
>
> A report exists and concludes that operational sequence analyses (OSAs) were conducted in conformance with the task analysis implementation plan. OSAs performed include the following:
>
> – Plant heatup and startup from post-refueling to 100 percent power
>
> – Reactor trip, turbine trip, and safety injection
>
> – Natural circulation cooldown (startup feedwater with steam generator)
>
> – Loss of reactor or secondary coolant

– Post-loss-of-coolant accident (LOCA) cooldown and depressurization

– Loss of RCS inventory during shutdown

– Loss of the normal residual heat removal system (RNS) during shutdown

– Manual automatic depressurization system (ADS) actuation

– Manual reactor trip via the protection and monitoring system (PMS), via diverse actuation system (DAS)

– ADS valve testing during mode 1

In DCD Revision 17, the applicant deleted this ITAAC because it had completed the work described.

18.5.9.2 Evaluation

The task analysis consists of a function-based task analysis and two OSA analyses (OSA-1 and OSA-2). As documented in its safety evaluation of the AP1000 DCD Revision 15, the staff reviewed the function-based task analysis and OSA-1 results and concluded that these task analyses are complete. As part of the DCD Revision 17 review, the staff reviewed the OSA-2 Implementation Plan and OSA-2 Summary Report. The reports describe the detailed methodology the applicant used to conduct OSA-2, as well as the results and impact on the four issues described in the OSA-2 Implementation Plan: 1) completeness of available information; 2) time to perform tasks; 3) operator workload analysis; and 4) operational crew staffing. As described, the task analysis was used in establishing the basis for the HFE design.

The staff reviewed the OSA-2 Summary Report (APP-OCS-J1R-220), Revision A, which provides the results of OSA-2 for the AP1000 design in accordance with the implementation plan. The implementation plan summarizes [ ] components and the corresponding [ ] MTIS tasks for analysis. The results summary report also describes [ ] scenarios and [ ] associated tasks that were described in the implementation plan and analyzed during OSA-2 implementation. Open Item OI-SRP18-COLP-02A documented that the task analysis had not been completed for all of the MTIS tasks. This work was subsequently completed and submitted for staff review in OSA-2 Summary Report (APP-OCS-J1R-220), Revision B. Revision B of the results summary report includes the following information:

(1) The report summarizes the analysis of the [ ] risk-important MTIS tasks.

Westinghouse includes a description of the [ ] and the task analysis results for the MTIS in its results summary report. [ ] similar to OSA, which provides a [ ]. This analysis uses "[ ]" logic, which enables the evaluator to determine [ ] for the MTIS tasks. This is consistent with the specification in Criterion 2 (in Section 5.4 of NUREG-0711) that the applicant uses a process like OSA. Appendix C to the summary report presents the results of the MTIS analyses.

Section 2.1 of the summary report discusses the [ ] scenarios developed as a basis for the total of [ ] tasks to be analyzed using the OSA-2 methodology. (The [ ] tasks equate to [ ] scenarios because [ ].) For each scenario, the description in Appendix A to the summary report includes the [ ]. Appendix B to the summary report discusses the results of the analyses.

The summary report briefly describes the OSA-2 analyses of these [ ] scenarios and [ ] tasks. The analyses identified [ ] risk-important tasks and the following [ ].

The OSA-2 Summary Report contains tables giving detailed [ ], as well as the [ ]. The descriptions include the [ ]. For example, each scenario and its related tasks are labeled as “[ ]. In the case of an [ ], the first basic event is [ ]. The beginning step or cue in this case is the [ ]. The second cue is that at [ ]. Westinghouse continues to describe the next few events including the [ ]. This analysis continues until the [ ]. Westinghouse’s OSA-2 for this particular task includes [ ]. In this case, each task associated with the basic event [ ].

The staff has concluded that the task analysis for Revision 17 of the AP1000 DCD is complete and has sufficient depth to support control room inventory identification and workload analysis. Open Item OI-SRP18-COLP-02A is closed and Design Commitment 2 in ITAAC Table 3.2-1 (DCD Revision 15) is complete and closed.

18.5.9.3 Conclusion

The staff has reviewed the OSA-2 Implementation Plan, Revision 1, and the OSA-2 Summary Report, Revision B, and has determined that the applicant has adequately addressed the criteria found in Section 5 of NUREG-0711. In addition, the staff’s review has determined that there is sufficient information to close COL Information Item 18.5-2 (NUREG-1793 Item 18.5.3-1), COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-2) and COL Information Item 18.5-1 (NUREG-1793 Item 18.5.3-3). Design Commitment 2 in ITAAC Table 3.2-1 (DCD Revision 15) is complete and closed. The task analysis that was completed under this ITAAC provides reasonable assurance that a complete Control Room Inventory has been identified. The task analysis also demonstrates that the HFE design ensures an acceptable workload for the operators.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.7 Element 6: Human Reliability Analysis

The applicant made no substantive changes to this section. However, Westinghouse submitted TR-59 (APP-GW-GLR-011, Revision 0, “AP1000 Standard Combined License Technical Report, Execution and Documentation of the Human Reliability Analysis/Human Factors Engineering Integration”) to close COL Information Item 18.7-1.

18.7.5 Evaluation of COL Information Item 18.7-1

18.7.5.1 Summary of Technical Information

COL Information Item 18.7-1 states the following:

> Combined license applicants referencing the AP1000 certified design will address the execution and documentation of the human reliability analysis/human factors engineering integration implementation plan that is presented in Section 18.7.

Westinghouse submitted TR-59 to close COL Information Item 18.7-1. TR-59 summarizes the applicant's method for conducting the HRA/HFE evaluation for the AP1000 and unites the relevant HRA/HFE evaluation implementation plan with the results documentation.

The staff reviewed and approved Westinghouse Commercial Atomic Power (WCAP)-14651, Revision 2, "Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan," as a supporting document for DCD Revision 15. Sections 2 through 5 of WCAP-14651 describe the major aspects of the plan:

- Section 2 discusses the PRA/HRA identification of critical HAs and risk-important tasks.
- Section 3 describes the task analyses for critical HAs and risk-important tasks.
- Section 4 discusses the reexamination of critical HAs and risk-important tasks.
- Section 5 provides information on the validation of HRA performance assumptions.

The staff used this implementation plan (in addition to NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 18, "Human Factors Engineering," Revision 2, issued March 2007, and NUREG-0711, Revision 2) to review WCAP-16555, Revision 1, "AP1000 Identification of Critical Human Actions and Risk Important Tasks." In addition to TR-59, Westinghouse provided WCAP-16555 to the NRC to close COL Information Item 18.7-1. In WCAP-16555, the applicant provided the results of the evaluation of the AP1000 PRA/HRA that identifies the critical HAs and risk-important tasks for plant operation.

18.7.5.2 Evaluation

The staff determined that WCAP-16555 addresses Section 2 of the WCAP-14651 implementation plan. The applicant addressed Sections 3 through 5 of the implementation plan in Parts 1 and 2 of the OSA.

Section 2 of WCAP-14651 relates to Criterion 1 in Section 7.4 of NUREG-0711, which states the following:

> Risk-important human actions should be identified from the PRA/HRA and used as input to the HFE design effort.
>
> - These actions should be developed from the Level 1 (core damage) PRA and Level 2 (release from containment) PRA including both internal and external events. They should be developed using selected (more than one) importance measures and HRA sensitivity analyses to provide

reasonable assurance that an important action is not overlooked because of the selection of the measure or the use of a particular assumption in the analysis.

Section 2 of WCAP-14651 discusses the PRA/HFE identification of critical HAs and risk-important tasks. Sections 2.1 and 2.2, respectively, describe the process used to identify critical HAs and risk-important tasks. Section 2.2 is divided into three sections describing the process to identify the risk-important quantitative, qualitative, and qualitative MTIS criteria.

Evaluation Criterion 1—Critical Human Actions: Section 2.1 of WCAP-14651 states that the applicant will determine critical HAs using both deterministic and PRA criteria. In Section 3.1 of WCAP-16555, the applicant presented the results of the analyses, which determined that there were no critical actions for the AP1000. For the deterministic criterion, there were no Type A (as defined in Sections 7.5.2.1 and 7.5.3.1 of the DCD) post-accident instruments and no HAs were required to mitigate any design-basis accident. For the PRA criteria, the analysis showed that no HA, when failed in the PRA, results in a core damage frequency (CDF) of 1×10^{-4} core damage events per reactor-year or greater. Further, no HA, when failed in the PRA, results in a large release frequency of 1×10^{-5} events per reactor-year. Thus, there are no critical actions for the AP1000 plant. This is in accordance with the design objectives of the AP1000.

Evaluation Criterion 2—Quantitative and Qualitative Risk-Importance Criteria: Section 2.2 of WCAP-14651 states that the applicant will use both quantitative and qualitative criteria to identify the risk-important tasks of the AP1000 design. The quantitative criteria are a risk achievement worth (RAW) of 3.0 and a risk reduction worth (RRW) of 1.1. The RAW is a value that examines the increase in risk that would result if a single HA were to fail. The RRW value examines the decrease in risk that would result if an HA were made perfectly reliable for a given process or parameter. The focused PRA reduced these values to an RAW of 2.0 and an RRW of 1.05.

Section 3.2 of WCAP-16555 and related tables provide the results of the evaluation using the RAW and RRW measures. The applicant performed evaluations for both the CDF and the large early release frequency and considered the internal events, flooding, fire, and shutdown PRAs. The applicant identified about 20 risk-important tasks, summarized in Table 3.2-2. The staff also compared the HAs in the dominant sequences with the top operator actions determined by the risk-importance measures. The dominant sequences and the operator actions were consistent. The staff finds that the applicant's use of quantitative risk-importance criteria meets the objective of the implementation plan.

Section 2.2 of WCAP-14651 includes five qualitative criteria for identifying additional risk-important tasks in conjunction with an expert panel. The applicant used the criteria listed in WCAP-16555, Section 2.2.1, to identify the qualitative risk-important HAs. These criteria are consistent with those in the implementation plan. The applicant also provided the results of this evaluation in Sections 2.2.1 and 3.2.1 of WCAP-16555. The expert panel identified three HAs that were added to the list of risk-important tasks. This approach to identifying the qualitative risk-important HAs is consistent with that given in the implementation plan. The staff finds this to be acceptable.

Evaluation Criterion 3—MTIS Risk-Importance Qualitative Criteria: Section 2.2 of WCAP-14651 provides qualitative criteria for identifying risk-important MTISs.

In Section 3.3 of WCAP-16555, the applicant gives the methodology used to identify the MTIS activities for the risk-important structures, systems, and components (SSCs). A group of engineers representing various disciplines and backgrounds, including HFE, HRA, and PRA, reviewed the results produced by this methodology. The applicant also provided Tables 3.3-1 and 3.3-2, which present the results of the MTIS evaluation. Table 3.3-1 includes the initial list of SSCs considered for MTIS activities, along with any other components that may be risk-important and have interfaces with the control room but may not have been included in the initial list. Lastly, Table 3.3-2 lists the representative MTIS activities that will receive the HFE review. In cases where the same MTIS activity was repeated for different SSCs, one of those MTIS activities from that list was chosen to represent (or selected as a “representative” of) that group.

The staff requested clarification in RAI TR-59-11 about the activities outside of the control room and whether they were included in the set of MTIS tasks identified through the expert panel. The staff noted that the Davis-Besse reactor vessel incident is an example of the need for proper MTIS task identification. The reactor vessel is a risk important SSC, and inspection of the vessel exterior would be an MTIS activity that seems worthy of appropriate planning at the design stage to address human factors issues associated with this activity. Thus, by including activities outside of the control room, accessibility can be assured and procedures and training provided to avoid the kinds of problems that occurred with reactor vessel leakage and corrosion. In its response dated July 27, 2007, the applicant provided information clarifying that operator actions outside of the control room were considered and noted that two of the actions considered were outside of the control room. Further, the passive nature of the plant design limits the use of manual control valves, and the manual control valves that are risk-important have main control room position indication.

The staff finds that the applicant has acceptably implemented the process specified in WCAP-14651 to identify the MTIS risk-important tasks.

Criterion 2 in Section 7.4 of NUREG-0711 states the following:

> Risk-important HAs and their associated tasks and scenarios should be specifically addressed during function allocation analyses, task analyses, HSI design, procedure development, and training. This will help verify that these tasks are well supported by the design and within acceptable human performance capabilities (e.g., within time and workload requirements).

Criterion 2 Evaluation

WCAP-14651, Section 3, describes the process for including the HRA risk-important activities in the task analysis. Westinghouse’s OSA documents (for OSA-1 and OSA-2) summarize how the applicant input the HRA risk-important tasks into the task analysis. The OSA-1 Summary Report, Table 3-1, specifically addresses the risk-important tasks. The OSA also detailed task sequences and performance requirements. The applicant gave details of its methodology for task identification with regard to emergency operating procedures (EOPs), system operating procedures, and general operating procedures (GOPs). Section 4.2.4 of the OSA-1 Summary Report presents recommendations for the risk-important actions. Finally, in Section 1 of the OSA-1 Summary Report, Westinghouse stated that the results of the OSA are a set of requirements and constraints on operator task performance and that these are fed into the HSI design. The staff finds that the applicant has acceptably implemented the process described in the implementation plan.

Criterion 3 in Section 7.4 of NUREG-0711 states the following:

> The use of PRA/HRA results by the HFE design team should be specifically addressed; that is, how are risk-important HAs addressed (through HSI design, procedural development, and training) under the HFE program to minimize the likelihood of operator error and provide for error detection and recovery capability.

The applicant submitted the implementation methodology for OSA-2 to address part of Sections 3 and 4 in WCAP-14651. The applicant also provided the OSA-2 Summary Report for review. These documents meet the objectives of Sections 3 and 4 of WCAP-14651, by assigning focus areas for operators, by including MTIS activities in OSA-2, and by using operating procedures during the process. The staff finds that the applicant acceptably implemented the process described in the implementation plan.

Criterion 4 in Section 7.4 of NUREG-0711 states the following:

> HRA assumptions such as decision making and diagnosis strategies for dominant sequences should be validated by walkthrough analyses with personnel with operational experience using a plant-specific control room mockup or simulator. Reviews should be conducted before the final quantification stage of the PRA.

WCAP-14651 describes the process for the validation of the HRA performance assumptions. The applicant is implementing this process as part of its integrated system validation for the AP1000. Section 10.0 of this report details the review of the process used to integrate the HRA risk-important HAs.

18.7.5.3 Conclusion

The staff concludes that TR-59 (APP-GW-GLR-011), WCAP-16555, and the related RAI response (RAI-TR59-11) describe an acceptable approach to implementing WCAP-14651 and to meeting the criteria in Section A.6 of NUREG-0800 and Section 7.4 of NUREG-0711. Based on this material COL Information Item 18.7-1 is closed.

18.7.6 Evaluation of Tier 1 Information—Design Commitment 1, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15)

18.7.6.1 Summary of Technical Information

ITAAC Design Commitment 1 reads as follows:

> Design Commitment: The integration of HRA with HFE design is performed in accordance with the implementation plan.
>
> Inspection, Tests, and Analyses: The applicant will perform an evaluation of the implementation for the integration of HRA with HFE design.

Acceptance Criteria: A report exists and concludes that critical human actions (if any) and risk important tasks were identified and examined by task analysis, and used as input to the HSI design, procedure development, staffing, and training.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.7.6.2 Evaluation

This ITAAC was deleted in Revision 16 of the AP1000 DCD (but the number was kept as a place holder), then subsequently removed entirely from Revision 17. For Revision 17 to the DCD, the applicant has provided the methodology and summary reports that show the risk-important tasks were examined and would have input into the other HFE elements listed in the acceptance criteria. Also, the work products provided by the applicant demonstrate the following:

- There are no “critical human actions” because of the AP1000 passive design.
- “Risk-important actions” as well as “significant” actions are identified and included in the HFE design process in accordance with NUREG-0711 guidance.
- The OSA-1 analysis included all identified actions from the HRA. OSA-2 is a reiterative analysis (see Section 18.5 of this report) that also includes input from the HRA.

18.7.6.3 Conclusion

The staff concludes that Design Commitment 1 in ITAAC Table 3.2-1 (DCD Revision 15) is complete and closed and the COL Information Item 18.7-1 is complete and closed because risk-important HAs have been identified in accordance with the implementation plan and these HAs have been appropriately implemented in the HFE design via the task analysis in OSA-2.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.8 Element 7: Human-System Interface Design

18.8.3 General Human System Interface Design Feature Selection

18.8.3.1 Summary of Technical Information

In DCD Section 18.8.1.8, the applicant deleted reference to the use of computer-based models of cognitive response to control room events as an analytic method supporting workload analysis. The applicant substituted the term “task analysis”: the sentence now reads, “Analytic methods include the use of task analysis.”

18.8.3.2 Evaluation

NUREG-1793 Section 18.8.1.3 discusses task analysis only from a generic perspective as one of the NUREG-0711 elements. NUREG-1793 Section 18.8 does not include specific methods for evaluating workload. In both cases, the change described above does not affect the evaluation or conclusions from this section of the safety evaluation. Section 18.5, "Task Analysis," provides an evaluation of the impact of the change on task analysis.

18.8.3.3 Conclusion

The staff concludes that this change does not affect the evaluation or results documented in NUREG-1793 Section 18.8.1.3.

18.8.4 Evaluation of COL Information Item 18.8-1

18.8.4.1 Summary of Technical Information

COL Information Item 18.8-1 states the following:

> The COL applicant referencing the AP1000 certified design is responsible for the execution and documentation of the HSI design implementation plan.

The applicant issued TR-82 to address this COL information item. In this document, the applicant stated that the COL item has been fully addressed and no additional work is required by the COL applicant.

18.8.4.2 Evaluation

The applicant has satisfactorily completed documentation of the HSI design implementation plan. The staff reviewed the completed documents referenced in TR-82 and concluded that they appropriately execute the HSI design implementation plan, as described in the AP1000 DCD, Revision 15. The specificity of design requirements clearly increased in the transition from the functional design level to design specifications. The documents were consistently clear across this procedural hierarchy. The scope of and specificity in the design documents provide reasonable assurance that the design process will effectively produce the design document needed to support procurement, construction and inspection activities.

This COL information item is redundant to Design Commitment 3 from ITAAC Table 3.2-1 (DCD Revision 16), which states that the HSI design is performed for the operation and control system in accordance with the HSI design implementation plan. Based on this redundancy, the COL information item is closed.

18.8.4.3 Conclusion

The applicant is completing design documents in accordance with the HSI design implementation plan. While the applicant has not completed execution of the HSI design implementation plan, the COL information item is being closed because it is redundant to an existing ITAAC.

18.8.5 Review of Human Factors Evaluation Style Guide (APP-OCS-J1-002) against NUREG-0711 Criteria

18.8.5.1 Summary of Technical Information

The applicant submitted AP1000 HSI Design Guidelines (APP-OCS-J1-002, Revision 0). This document implements several NUREG-0711 criteria that have not been previously reviewed at the implementation plan level. The evaluation below verifies that the AP1000 HSI Design Guidelines effectively address applicable NUREG-0711 criteria.

18.8.5.2 Evaluation

Criterion 1—Style Guide

NUREG-0711, Section 8.4.5, “HSI Detailed Design and Integration Criteria,” Criterion 1 states the following:

> Design-specific HFE design guidance (style guide) should be developed. The design of the HSI features, layout, and environment should incorporate HFE guidelines.

In APP-OCS-J1-002, the applicant provided a detailed set of HFE requirements for all HSIs similar to the level of detail in NUREG-0700, Revision 2, “Human-System Interface Design Review Guidelines,” issued May 2002. The goal of the document is to ensure that the AP1000 designs comply with applicable HFE design principles.

The staff concludes that this document meets this criterion for design-specific HFE guidance.

Subcriterion—Style Guide Content

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The content of the style guide should be derived from (1) the application of generic HFE guidance to the specific application, and (2) the development of the applicant’s own guidelines based upon design-related analyses and experience. The applicant may justify guidelines that are not derived from generic HFE guidelines based on an analysis of recent literature, analysis of current industry practices and operational experience, tradeoff studies and analyses, and the results of design engineering experiments and evaluations. The guidance should reflect the applicant’s design decisions that address the specific goals and needs of the HSI design.

In APP-OCS-J1-002, the applicant included a list of technical references used to develop specific HFE guidance for the AP1000 design. The applicant used NUREG-0700 as a major source. The following references also support the AP1000 HFE design guidance:

- [ ]
- [ ]
- [ ]

- [ ]
- [ ]
- [ ]
- [ ]

The staff concludes that these technical references represent a diverse and thorough set of inputs for the AP1000 guidance. The AP1000 design guidance includes design principles and specific design criteria for all of the AP1000 HSIs.

Subcriterion—Scope and Level of Detail

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The topics in the style guide should address the scope of HSIs included in the design and address the form, function, and operation of the HSIs, as well as environmental characteristics relevant to human performance.

In APP-OCS-J1-002, Section 3, the applicant described the scope of the design guidelines. This includes the MCR, remote shutdown station, and TSC. Specific HSI interfaces include the plant information system, alarm system, computerized procedures, safety systems, soft controls, dedicated controls, DAS, and large screen displays. The scope addresses all areas described by the previously reviewed program-level documents. APP-OCS-J1-002, Section 26, includes environment-related criteria.

The staff concludes that the design guideline addresses the HSI scope satisfactorily. The level of detail is consistent with that found in NUREG-0700, an accepted program for HFE design criteria.

Subcriterion—Guideline Specificity

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The individual guidelines should be expressed in concrete, easily observable terms. In general, generic HFE guidelines should not be used in their abstract form. Such generic guidance should be translated into more specific design guidelines that can, as much as possible, provide unambiguous guidance to designers and evaluators. They should be detailed enough to permit their use by design personnel to achieve a consistent and verifiable design that meets the applicant’s guideline.

The level of detail provided in individual guidelines is consistent with the specificity in NUREG-0700. In general, the guidelines provide quantifiable direction. For many of the guidelines, and particularly for those cases in which more general direction is given, the basis for the guideline is included. This reference provides direction on guideline implementation. The guidance is divided into required and optional categories, which provides additional support to the designers and evaluators.

The staff concludes that the direction provided in the design guidance document is of sufficient detail that design personnel will be able to achieve a consistent and verifiable design.

Subcriterion—Style Guide Ease of Use

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The style guide should provide procedures for determining where and how HFE guidance is to be used in the overall design process. The style guide should be written so that designers can readily understand it. The style guide should support the interpretation and comprehension of design guidance by supplementing text with graphical examples, figures, and tables.

APP-OCS-J1-002 provides generic direction stating that the design guidance will be used during the design process and to facilitate design verification. Implementation plans for both of these activities refer to the use of the [ ]. The plans cross-reference between [ ] and the applicable sections of APP-OCS-J1-002, which will likely facilitate the use of the [ ], as indicated in the criterion, [ ], and the [ ] is provided to answer questions that might arise as to the applicability of the design guidance.

The staff concludes that the design guidance in APP-OCS-J1-002 is presented in a manner likely to facilitate its use by designers and evaluators. The applicant has provided sufficient cross-referencing in procedures to ensure their appropriate use.

Subcriterion—Usability

NUREG-0711, Section 8.4.5, Criterion 1, states the following:

> The guidance should be maintained in a form that is readily accessible and usable by designers and that facilitates modification when the contents require updating as the design matures. Each guideline included in the guidance documentation should include a reference to the source upon which it is based.

The applicant maintains APP-OCS-J1-002 on its electronic document tracking system as a controlled document. This ensures document accessibility and facilitates usability by virtue of word search capability. The document itself is [ ]. The applicant has demonstrated the ability to keep the document updated by incorporating more detail on [ ]. Each guideline includes a reference to source material; this should also aid the designer in determining how best to implement the requirements and to facilitate the evaluation of tradeoffs.

18.8.5.3 Conclusion

The staff concludes that APP-OCS-J1-002 provides specific HFE design guidance that satisfactorily implements NUREG-0711 criteria. The document provides sufficient detail to ensure that the process is consistently followed and provides reasonable assurance that design requirements are properly factored into the HSIs.

18.9 Element 8: Procedure Development

The applicant made no substantive changes to this section. However, Westinghouse submitted TR-70 (APP-GW-GLR-040, Revision 1, "Plant Operations, Surveillance, and Maintenance Procedures") to close COL Information Item 18.9-1.

18.9.5 Evaluation of COL Information Item 18.9-1

18.9.5.1 Summary of Technical Information

COL Information Item 18.9-1 was identified in NUREG-1793 and does not have a counterpart in the DCD. This COL action item is divided into two parts. The COL action item states the following:

> With regard to procedure development, the COL applicant will (1) address the procedure development considerations in NUREG-0711, and (2) identify the minimum documentation that the COL applicant will provide to the staff to complete its review.

Westinghouse submitted TR-70 for staff review. This report documents the methodology, criteria, and schedules for procedure development. The document addresses the information needed to close COL Information Item 18.9-1. The applicant made the TR-70 supporting documents available to the staff for the purpose of closing COL Information Item 18.9-1. Two of these documents were the writer's guides for normal operating procedures and two-column operating procedures (APP-GW-GJP-100, Revision G, "AP1000 Normal Operating Procedures (NOPs) Writer's Guideline," and APP-GW-GJP-200, Revision D, "Writer's Guideline for Two Column Procedures," respectively). The writer's guidelines explain the programmatic process that controls the preparation of the normal operating procedures and two column procedures.

The goal of the staff's review was to address each part of the action item. Consequently, the evaluation is described in two parts. Part 1 details how the applicant addressed the procedure development considerations in NUREG-0711. Part 2 describes the documents that were submitted to the staff for review.

18.9.5.2 Part 1—Evaluation

The staff reviewed TR-70 in combination with the writer's guides. The staff verified that the applicant had implemented the guidelines specified in WCAP-14690, "Designer's Input to Procedure Development for the AP600." WCAP-14690 is the staff-approved document that describes the methodology the COL applicant should use to develop procedures. In NUREG-1793, the staff approved the use of this document as a guide for procedures development and an acceptable guideline for creation of an implementation plan for the AP1000. In its review, the staff found that the writer's guides meet the criteria in NUREG-0711, Section 9.4, for the basis, development, and content of the AP1000 two column and normal operating procedures. The staff found that the information in TR-70 is consistent with the guidelines in WCAP-14690. Section 2.0 of WCAP-14690 details the general criteria that an applicant should implement to develop procedures. TR-70 addresses all of the guidance criteria in Section 2.0 of WCAP-14690. Section 4.0 of WCAP-14690 provides guidance on the process that should be used to write the plant-specific EOPs. Sections 3.0 and 5.0 of the WCAP describe the guidance for creation of the implementation plan with regard to computer-based procedures (CBPs). The following section documents the CBP evaluation as a subpart to addressing Part 1 of COL Information Item 18.9-1.

Human Factors Engineering Aspects of Computer-Based Procedures

The applicant did not address the impact of computerized procedures and accessibility in the original DC application. In the staff's evaluation of the AP1000 DCD, NUREG-1793 states the following:

> Evaluation of the applicant's computerized procedure system was not included in the design certification for the AP1000. WCAP-14690, Revision 1, provides information on the computer-based procedure system which will serve as the interface to the plant procedures.

NUREG-0700, Section 8; Interim Staff Guidance (ISG)-05 ("Task Working Group #5: Highly-Integrated Control Room—Human Factors Issues"); and NUREG-0711, Section 9.4, Criteria 7 and 9, are used to evaluate the methodology used to design the CBP system and the interaction between the operator and that system. ISG-05 is used as complementary review guidance for Criterion 9.

Criterion 7 states the following:

> An analysis should be conducted to determine the impact of providing CBPs and to specify where such an approach would improve procedure utilization and reduce operating crew errors related to procedure use. The justifiable use of CBPs over paper procedures should be documented. An analysis of alternatives in the event of loss of CBPs should be performed and documented.

In TR-70 or in the supporting referenced documentation, the applicant addressed the impact and utilization of CBPs not addressed in the original DC application. In Section 2.7 of TR-70, Revision 1, the applicant stated that comments from operations personnel involved in the human factors testing of the AP1000 control room design, and specifically the computerized procedure system, have been generally favorable. The applicant also documented the results of the analysis of the impact of providing CBPs in the referenced report WCAP-14645-NP, Revision 3, "Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant." The staff reviewed WCAP-14645-NP, Revision 3. The applicant identified multiple human performance issues with the CBPs and then noted the solution, or proposed solution, for each issue.

The staff issued RAI-SRP18-COLP-14 to the applicant requesting the analysis of alternatives to CBPs, in the event that a loss of CBPs occurs. In the RAI-SRP18-COLP-14 response dated August 4, 2008, the applicant stated that it would conduct this analysis as part of the second OSA, described in Section 2.1 of APP-OCS-J1R-210.

Subsequent to this RAI, the staff reviewed APP-OCS-J1R-220, Revision B, OSA-2 Summary Report. The OSA-2 Summary Report identifies the [ ]. This task, [ ], has [ ] that are described in Scenario 16. Also, in this section of the summary report, Westinghouse described how the [ ]. Appendix B, Section B.22, to the report gives details of the [ ] steps described in Scenario 16.

Open Item OI-SRP-COLP-19 was established to track an RAI clarifying how a loss of CBPs is managed. In the RAI-SRP18-COLP-19 response, dated September 1, 2009, the applicant provided the staff with this clarification:

- [ ].
- [ ].
- [ ].

Based on this information the open item was closed.

The staff conducted an audit of the CBP interface at the Westinghouse Energy Center in Monroeville, Pennsylvania in September 2009. During the audit the staff reviewed the AP1000 Computerized Procedure System (CPS) design process, including supporting documentation, as well as the characteristics and functions of the current system as implemented in the AP1000 engineering test simulator. The CPS characteristics and functions included [ ].

Based on the audit the staff concluded that the Westinghouse AP1000 CPS system was designed in accordance with the NRC certified HSI Design Implementation Plan and that all supporting documentation was acceptable and consistent with the NRC design review guidance including the guidance specific to CBP systems. The design as currently implemented is consistent with Westinghouse's design procedures and documentation.

Criterion 9 states the following:

> The physical means by which operators access and use procedures, especially during operational events, should be evaluated as part of the HFE design process. This criterion generally applies to both hard-copy and computer-based procedures, although the nature of the issues differs somewhat depending on the implementation.

The staff used ISG-05 as the complementing review guidance for NUREG-0711, Section 9.4, Criterion 9. ISG-05 provides review criteria for how the user will interface with the CBP system. The applicant provided the documentation to satisfy the ISG-05 criteria in APP-OCS-J1-020, Revision A, "Computerized Procedures System Functional Requirements." APP-OCS-J1-020 documents how the operator physically interfaces with the computer procedure system. The technical information in APP-OCS-J1-020 is consistent in addressing the criteria in ISG-05. CBPs are designed to be the primary procedure interface and access is gained via the video display units. Audit observations confirmed that the CBP system is easily accessed from visual display unit (VDU) menus. Navigation to a specific procedure is via a procedure menu. Navigation between procedures is typically driven by embedded links but the operator can also return to the main menu to select the desired procedure. Navigation was found to be simple and straightforward. Use of the hardcopy procedures, which are available in the control room as a backup to the CBPs, followed conventional practices. The staff submitted RAI-SRP18-COLP-11 requesting clarification of the CBP automation and whether the AP1000 computer procedure system would be computer-paced or user-paced. In its

RAI-SRP18-COLP-11 response, dated August 4, 2008, the applicant stated that this issue would not be of any consequence because the computer-paced function would be removed. The staff found this response acceptable.

18.9.5.3 Part 1—Conclusion

The staff determined that TR-70 and the writer's guides for normal and two column procedures together constitute an acceptable implementation plan for procedure development. This is because: (1) the documents address the criteria in the staff-approved WCAP-14690, which explains the process the procedure writer should take to develop an implementation plan; and (2) the documents also address the applicable criterion in the procedures development chapter in NUREG-0711.

The staff concludes that Westinghouse has designed a system that ensures the usability and usefulness of CBPs. Specifically, loss of the CBP HSI is appropriately addressed in procedures and training. Support provided for the transition to paper based procedures provides reasonable assurance that such a failure would not significantly impact the operator's ability to implement the appropriate accident response procedures. Further, the staff concluded that Westinghouse's approach for implementing a new technology into the control room and operating practices was acceptably conservative and should provide for a smooth transition to computerized operation of important procedures, such as EOPs. This approach will minimize any safety concerns associated with the loss of the CPS.

Based on the preceding information, the staff concludes that COL Information Item 18.9-1, Part 1 is complete and closed.

18.9.5.4 Part 2—Evaluation

To address the second part of COL Information Item 18.9-1, in addition to submitting TR-70, the applicant stated in Revision 17 of the AP1000 DCD that the COL applicant will be responsible for addressing the operational and programmatic issues and training to complete the AP1000 COL licensing process. Westinghouse would be responsible for managing the development, review, and approval of the AP1000 normal operating, abnormal operating, emergency operating, refueling and outage planning, alarm response, administrative, and MTIS procedures, as well as the procedures that address the operation of post-72-hour equipment.

18.9.5.5 Part 2—Conclusion

In DCD Tier 2, Revision 17, responsibility for completing this COL action was assumed by Westinghouse. As described above, sufficient documentation has been submitted to satisfy the criteria in NUREG-0711, Section 9.4. COL applicants have continuing responsibilities related to training and procedures but these are evaluated as part of operating program inspections. This Westinghouse response satisfies Part 2 of COL Information Item 18.9-1 (NUREG-1793 Item 18.9.3-1).

The staff concludes that the applicant's procedure development program provides reasonable assurance that procedures will support and guide human interaction with plant systems, as well as control plant-related events and activities. Human engineering principles and criteria are applied, along with all of the other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to use, validated, and in conformance with 10 CFR 50.34(f)(2)(ii). In addition, this closes OI-SRP18-COLP-19.

The staff concludes that COL Information Item 18.9-1, Part 2 is complete and closed. COL Information Item 13.5-1 covers the remainder of the procedures development.

18.11 Element 10: Human Factors Verification and Validation

Westinghouse submitted the following implementation plans to address COL Information Item 18.11-1 and ITAAC Design Commitment 4, Tier 1, Table 3.2-1 (DCD Revision 15):

- APP-OCS-GEH-120, "AP1000 Human Factors Engineering Design Verification Plan," Revision B
- APP-OCS-GEH-220, "AP1000 Human Factors Engineering Task Support Verification Plan," Revision B
- APP-OCS-GEH-320, "AP1000 Human Factors Engineering Integrated System Validation Plan," Revision D (Integrated System Validation (ISV) Plan)
- APP-OCS-GEH-321, "AP1000 Human Factors Engineering Integrated System Validation Scenario Information," Revision B (ISV Scenario Plan)
- APP-OCS-GEH-420, "Human Factors Engineering Discrepancy Resolution Process," Revision B
- APP-OCS-GEH-520, "AP1000 Human Factors Engineering Design Verification at Plant Startup," Revision B

NUREG-0711 states the following:

> An implementation plan gives the applicant's proposed methodology for meeting the acceptance criteria of the element. An implementation plan review gives the applicant the opportunity to obtain staff review of and concurrence in the applicant's approach before conducting the activities associated with the element. Such a review is desirable from the staff's perspective because it provides the opportunity to resolve methodological issues and provide input early in the analysis or design process when staff concerns can more easily be addressed than when the effort is completed.

The staff will verify the final results of the design analyses to ensure that the design is completed in accordance with the process specified in the implementation plans in accordance with the design acceptance criteria (DAC) approach. This may occur via a DC amendment, the COL application review, or through the ITAAC closure process.

When conducting an implementation plan review, the staff needs to:

- understand how the detailed methodology will be implemented
- determine that the methodology can be reliably conducted by design personnel
- be confident that the methodology will provide results that will be acceptable as evaluated by the relevant NUREG-0711 review criteria

18.11.5 Evaluation of COL Information Item 18.11-1

18.11.5.1 Summary of Technical Information

COL Information Item 18.11-1 states the following:

> Combined License applicants referencing the AP1000 certified design will address the development, execution and documentation of an implementation plan for the verification and validation of the AP1000 Human Factors Engineering Program. The programmatic level description of the AP1000 verification and validation program presented and referenced by Section 18.11 will be used by the Combined License applicant to develop the implementation plan.

18.11.5.2 Evaluation

COL Information Item 18.11-1 includes two distinct activities related to the AP1000 HFE program V&V. The first activity addresses development of an implementation plan. Design Commitment 4, Tier 1, Chapter 3, ITAAC Table 3.2-1, of the AP1000 DCD, Revision 15, also addresses this commitment. The second activity is to execute and document the execution of the implementation plan. Design Commitment 5, Tier 1, Chapter 3, ITAAC Table 3.2-1, of the AP1000 DCD, Revision 15, addresses this commitment.

18.11.5.3 Conclusion

The NRC staff notes that COL Information Item 18.11-1 is similar to existing Design Commitments 4 and 5, ITAAC Table 3.2-1 (DCD Revision 15). The development of the implementation plans has been completed and these implementation plans are evaluated below under Evaluation of Tier 1 Information Design Commitment 4 below. The execution and documentation of the implementation plans will be addressed in Design Commitment 5, ITAAC Table 3.2-1. Thus, COL Information Item 18.11-1 is no longer needed since the work has either been completed by Westinghouse or will be completed under the DCD ITAAC 5.

18.11.6 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 1 of 5, HSI Task Support Verification

18.11.6.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 1): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the …HSI task support verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.6.2 Evaluation

NUREG-0711, Section 11.4.2.2, Criterion 1, states the following:

> The criteria for task support verification come from task analyses of HSI requirements for performance of personnel tasks.

Evaluation of Criterion 1

In APP-OCS-GEH-220, the applicant provided a specific verification plan for each of the task analysis inputs as outlined below:

- Section 2.2 is the verification plan for the function based task analysis. APP-OCS-J1A-030, Revision A, "FBTA Summary Report," provides [ ]. The [ ]. If the final design does not implement the recommendations, [ ].

- Section 2.3 is the verification plan for OSA-1. These tasks are derived from [ ]. A database is used to maintain the tasks identified by this analysis. Before final task verification, the plan requires the database to be [ ]. The independent verifier ensures that for each unique operator action, [ ].

- Section 2.4 is the verification plan for OSA-2 [ ]. If a new task is identified, then the OCS product manager ensures that disposition of the task is addressed. For each task identified in OSA-2, a list of [ ]. The independent verifier then confirms that the HSI resource is available, the HSI display information is appropriate, the communication facility is available and located appropriately, and the labeling is correct.

- Section 2.5 is the verification plan for the OSA-2 tasks specific to [ ]. Verification follows the same process as that used for OSA-1.

- Section 2.6 is the verification plan for the OSA-2 tasks specific to [ ]. Verification follows the same process as that used for OSA-1.

The staff concludes that APP-OCS-GEH-220 provides clear, specific direction on how the results of each specific task analysis are verified. Acceptance criteria are stated within the procedure and, when combined with the use of an independent verifier, provide reasonable assurance that the HSI requirements properly incorporate the task analysis results.

NUREG-0711, Section 11.4.2.2, Criterion 2, "General Methodology," states the following:

> The HSIs and their characteristics (as defined in the HSI inventory and characterization) should be compared to the personnel task requirements identified in the task analysis.

Evaluation of Criterion 2

The implementation plan for task support verification, as outlined above, provides clear direction that the final HFE design is to be compared to personnel task requirements. Direction is provided to document and justify or resolve all deviations. The direction is structured so that each task is specifically addressed. This supports a clear communication of source documents and acceptance criteria to be used in the verification.

The staff concludes that APP-OCS-GEH-220 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for the general methodology of task verification.

NUREG-0711, Section 11.4.2.2, Criterion 3, states the following:

> Human engineering discrepancies (HEDs) should be identified when an HSI needed for task performance is not available or when HSI characteristics do not match personnel task requirements.

Evaluation of Criterion 3

In APP-OCS-GEH-220, the applicant stated that any time an HSI resource or an appropriate display is not available; a discrepancy worksheet is filled out. The procedure specifically states the following verification points:

- [ ].
- [ ].
- [ ].
- [ ].
- [ ].

When the V&V evaluation is complete, the OCS product manager assesses each work discrepancy worksheet. Discrepancies that are directly justified as exceptions are not considered HEDs. The applicant documents justified discrepancies as part of future report APP-OCS-GER-120, “AP1000 HFE Task Support Verification Report,” along with a list of HEDs identified by discrepancy reports. APP-OCS-GEH-420 provides an implementation plan for resolving the discrepancy worksheets that are not justified by the product manager.

The staff concludes that APP-OCS-GEH-220 provides sufficient details of the implementation plan to satisfactorily demonstrate implementation of this NUREG criterion for identifying task requirement deficiencies during task verification.

NUREG-0711, Section 11.4.2.2, Criterion 4, states the following:

> An HED should be identified for HSIs that are available in the HSI, but are not needed for any task.…

Evaluation of Criterion 4

In APP-OCS-GEH-220, Sections 2.3.2 (OSA-1) and 2.4.2 (OSA-2), the applicant stated that the independent verifier will check each display for information and/or controls that are not associated with task requirements. Deviations must be documented on a discrepancy worksheet.

The staff concludes that APP-OCS-GEH-220 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for identifying unnecessary HSI components during task verification.

18.11.6.3 Conclusion

The staff concludes that APP-OCS-GEH-220 provides an implementation plan that satisfactorily implements the guidance in NUREG-0711 relative to task support verification. The level of detail provided and the use of an independent verifier provides reasonable assurance that the HSI requirements properly incorporate the results from all task analyses performed. This element of ITAAC Design Commitment 4 (DCD Revision 15), as described above, is complete and closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.11.7 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 2 of 5, HFE Design Verification

18.11.7.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (Part 2): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the …HFE Design Verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.7.2 Evaluation

NUREG-0711, Section 11.4.2.3, Criterion 1, states the following:

> The HFE guidelines serve as review criteria. Selection of specific guidelines depends on the characteristics of the HSI components included in the scope of review and whether the applicant has developed a design-specific guideline document. NUREG-0700 may be used for HFE design verification.

Evaluation of Criterion 1

In APP-OCS-GEH-120, the applicant stated that HSI resources and operation and control centers are verified against APP-OCS-J1-002. APP-OCS-J1-002 satisfactorily implements NUREG-0711, Section 8.4.5(1), as described in Section 18.8. It includes guidance from NUREG-0700, Revision 2, and Commission Electrotechnique Internationale/International Electrotechnical Commission (CEI/IEC) 964, "Design for Control Rooms of Nuclear Power Plants," which program-level documents specifically cite. The report also includes results from operating experience review, function-based task analysis, and other industry guidance.

The staff concludes that APP-OCS-GEH-120 provides sufficient direction to ensure that the HFE guidelines serve as review criteria and have an appropriate level of detail. The report is also consistent with the program description.

NUREG-0711, Section 11.4.2.3, Criterion 2, states the following:

> The applicant should compare the characteristics of the HSI components with the HFE guidelines to determine whether the HSI is acceptable or discrepant (i.e., an HED).
>
> The applicant should evaluate discrepancies as potential indicators of additional issues.

Evaluation of Criterion 2

In APP-OCS-GEH-120 the applicant provided a complete list of [ ] (Section 1.2.2). The general process description in Section 2.1 specifies that each [ ] APP-OCS-J1-002. Appendices B and C provide [ ]. APP-OCS-J1-002 provides pass/fail criteria. A discrepancy worksheet documents all discrepancies. Disposition of discrepancies can be handled immediately by the OCS product manager or submitted to the AP1000 HFE engineering discrepancy resolution process described in APP-OCS-GEH-420. A future report, APP-OCS-GER-120, will describe all discrepancies and their justification or resolution.

The staff concludes that the implementation plan provides a disciplined process for verifying that the HSI design effectively implements design acceptance criteria. Discrepancies are documented and subjected to a corrective action process that evaluates the potential for additional issues. The staff concludes that APP-OCS-GEH-120 provides sufficient detail to satisfactorily demonstrate implementation of this NUREG criterion for design verification methodology.

NUREG-0711, Section 11.4.2.3, Criterion 3, states the following:

> The applicant should document HEDs in terms of the HSI component involved and explain how the characteristics depart from a particular guideline.

The evaluation of this criterion is contained in the evaluation of Criterion 2, directly above.

18.11.7.3 Conclusion

The staff concludes that APP-OCS-GEH-120 provides an implementation plan that satisfactorily implements the NUREG-0711 criteria associated with design verification. The document provides reasonable assurance that the HSI designs reflect the design requirements. This element of ITAAC Design Commitment 4 (DCD Revision 15), as described above, is complete and closed.

18.11.8 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 3 of 5, Integrated System Validation

18.11.8.1 Summary of Technical Information

At the time of the Westinghouse AP1000 DC, based on Revision 15 of the DCD, HFE V&V was reviewed and found acceptable at a programmatic level. The Westinghouse V&V program was described in WCAP-15860, "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan," Revision 2, dated October 2003. Pursuant to Section 18.11.1 of the AP1000 DCD, a COL applicant referencing the AP1000 is committed to developing an implementation plan for V&V consistent with the NRC approved programmatic description in WCAP-15860. ITAAC Design Commitment 4 (Tier 1 Section 3.2, Human Factors Engineering, Table 3.2-1) states:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 3): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the …Integrated System Validation activity.

To fulfill this commitment, Westinghouse has submitted WCAP-16769-P, "AP1000 Human Factors Engineering Verification and Validation," and two implementation plans:

- APP-OCS-GEH-320 (ISV Plan)
- APP-OCS-GEH-321 (ISV Scenario Plan)

Open Item OI-SRP18-COLP-03A was created by the staff to track the review of these documents.

18.11.8.2 Evaluation

The purpose of this review is to determine whether the applicant's ISV Plan and its companion document, the ISV Scenario Plan, provide an acceptable implementation plan in accordance with NUREG-0711. These documents are evaluated using WCAP-15860 and the NUREG-0711 review criteria for operational condition sampling and ISV.

18.11.8.2.1 Applicable Review Criteria

When the staff has an NRC-certified, programmatic-level description of an HFE activity, the review criteria used to evaluate an implementation plan come from two sources: the certified, programmatic description and NUREG-0711. The programmatic description, WCAP-15860, identifies the general ISV approaches and constraints. The staff's review of the ISV Plan's compliance with WCAP-15860 is discussed in Section 18.11.8.2.2 below.

NUREG-0711 criteria were used to evaluate the detailed methodology (taking into account the approved approach described in the WCAP). The ISV review criteria used were from the following sections of NUREG-0711:

- Section 11.4.1 - Operation Condition Sampling
  - Sampling Dimensions (3 review criteria)
  - Identification of Scenarios (2 review criteria)

- Section 11.4.3 - Integrated System Validation
  - Test Objectives (1 review criteria)
  - Validation Test Beds (9 review criteria)
  - Plant Personnel (4 review criteria)
  - Scenario Definition (3 review criteria)
  - Performance Measurement (5 review criteria)
  - Test Design (9 review criteria)
  - Data Analysis and Interpretation (5 review criteria)
  - Validation Conclusions (2 review criteria)

In this document, the NUREG-0711 criteria are used to assess the completeness of the ISV Plan and its acceptability as an implementation plan. The results of the staff's evaluation of the ISV Plan with respect to the NUREG-0711 criteria are provided in Sections 3 and 4 for operational condition sampling and ISV methodology, respectively.

18.11.8.2.2 Compliance with the WCAP-15860

The staff evaluated whether the ISV Plan was developed in accordance with the commitments made in WCAP-15860 and whether the ISV Plan satisfies the NRC review criteria of NUREG-0711, Section 11. In general, the ISV Plan follows the commitments made in WCAP-15860. Inconsistencies noted in earlier revisions of the ISV Plan were documented in RAI-22 and the specific details have now been acceptably addressed in Revision D of the ISV Plan.

Additionally, Section 1.5 of the ISV Plan now states that the ISV Plan conforms to the commitments, scope, purpose, and issues as stated in WCAP-15860, with the exception of two areas where exceptions have been taken. The staff has reviewed these two exceptions and found them acceptable for the reasons stated below.

Exception 1: WCAP-15860 states that ISV will utilize currently qualified operating crews as the participants. However, as AP1000 is a new plant design, the ISV participants will not be fully qualified and experienced AP1000 operators. The ISV subjects will not have the same task performance proficiency as that of fully qualified AP1000 operators.

Evaluation: The staff finds this exception acceptable because the ISV Plan continues to include operating experience specifications that are sufficient to ensure valid testing. The ISV will be relatively more demanding and thus a more conservative test of the HFE design. Section 18.11.8.2.4.3, "Plant Personnel," of this report provides additional detail.

Exception 2: WCAP-15860 states that ISV will address all of the EOPs. However, the ISV Plan states that the ISV scenarios will include a representative subset of the EOPs. The ISV scenarios will ensure that all functional operator knowledge, skills, and abilities addressed in the EOPs are assessed.

Evaluation: The staff finds this exception acceptable because:

- The applicable NUREG-0711 review criteria do not call for 100 percent coverage of procedures during ISV. Section 18.11.8.2.3, "Compliance with NUREG-0711 - Operational Conditions Sampling (OCS)," of this report provides additional detail.

- Westinghouse verified the following:

> The ISV scenarios will ensure that all functional operator knowledge, skills and abilities addressed in the AP1000 EOPs are examined and validated in ISV. While the ISV scenarios may not explicitly cause the operators to enter each of functional recovery procedures, the demand to perform similar EOP steps will be represented [ ]

Additionally, prior to the ISV, [ ]. It also ensures a thorough ISV process. Thus, this exception is acceptable.

18.11.8.2.3 Compliance with NUREG-0711 - Operational Conditions Sampling (OCS)

NUREG-0711, Section 11.4.1, states, "The sampling methodology will identify a range of operational conditions to guide V&V activities. The review of operational conditions sampling considers the dimensions to be used to identify and select conditions and their integration into scenarios."

The objective of reviewing operational condition sampling is to verify that the applicant has identified a sample of operational conditions that: (1) includes conditions that are representative of the range of events that could be encountered during operation of the plant; (2) reflects the characteristics that are expected to contribute to system performance variation; and (3) considers the safety significance of HSI components. These sample characteristics are best identified through the use of a multidimensional sampling strategy to provide reasonable assurance that variation along important dimensions is included in the V&V evaluations.

The staff reviewed the defined scenarios in the ISV Plan and the ISV Scenario Plan to determine whether the OCS dimensions were addressed. The aspects of the specified OCS, both from WCAP-15860 and from NUREG-0711 have been addressed by the ISV Plan and the ISV Scenario Plan.

18.11.8.2.3.1 Sampling Dimensions

The sampling dimensions addressed in NUREG-0711, Section 11.4.1.2, include plant conditions, personnel tasks, and situational factors known to challenge personnel performance.

(1) The following plant conditions should be included:

- normal operational events including plant startup, plant shutdown or refueling, and significant changes in operating power

- failure events, e.g.,
  - instrument failures [e.g., safety-related system logic and control unit, fault tolerant controller, local “field unit” for multiplexer (MUX) system, MUX controller, and break in MUX line] including instrumentation and control (I&C) failures that exceed the design basis, such as a common mode I&C failure during an accident

  - HSI failures (e.g., loss of processing and/or display capabilities for alarms, displays, controls, and computer-based procedures)

- transients and accidents, e.g.,
  - transients (e.g., turbine trip, loss of off-site power, station blackout, loss of all feedwater, loss of service water, loss of power to selected buses or MCR power supplies, and safety and relief valve transients)

  - accidents (e.g., main steam line break, positive reactivity addition, control rod insertion at power, anticipated transient without scram, and various-sized LOCAs)

  - reactor shutdown and cool down using the remote shutdown system

- reasonable, risk-significant, beyond-design-basis events, which should be determined from the plant-specific PRA

- consideration of the role of the equipment in achieving plant safety functions [as described in the plant safety analysis report (SAR)] and the degree of interconnection with other plant systems. A system that is interconnected with other systems could cause the failure of other systems because the initial failure could propagate over the connections. This consideration is especially important when assessing non-Class 1E electrical systems.

Evaluation of Criterion (1)

WCAP-15860, Section 4.6 includes an extensive and multi-dimensional set of criteria that address this particular criterion. The ISV Plan was developed based on this criterion and includes [ ] separate scenarios, which are detailed in the ISV Scenario Plan. ISV Plan, Section 5.1.1, “Events,” lists the various [ ] scenarios. Also, the ISV Scenario Plan has an Appendix A titled, “Scenario Specifications” that provides the [ ] scenarios. The scenarios themselves follow in the rest of Appendix A. Additionally, the ISV Scenario Plan, Appendix E, “Tasks of Special Interest,” lists each [ ] and in which scenario(s) it is addressed. Appendix E also provides a cross reference between scenarios and the [ ]. This scenario information was compared with the commitments of WCAP-15860 and with the NUREG-0711 criteria. The ISV Plan and ISV Scenario Plan together were found to satisfy both the programmatic plan and NUREG-0711 and meet the criterion on plant conditions.

(2) The following types of personnel tasks should be included:

- Risk-significant HAs, systems, and accident sequences - All risk-important HAs should be included in the sample. These include [those] identified in the PRA and those identified as risk-important in the SAR and NRC's SER. Situations where human monitoring of an automatic system is risk-important should be considered. Additional factors should be sampled that contribute highly to risk, as defined by the PRA, including:
  - dominant human actions (selected via sensitivity analyses)
  - dominant accident sequences
  - dominant systems (selected via PRA importance measures such as RAW or RRW)

- OER-identified difficult tasks—The sample should include all personnel tasks identified as problematic during the applicant's review of operating experience.

- Range of procedure guided tasks—These are tasks that are well defined by normal, abnormal, emergency, alarm response, and test procedures. The operator should be able to, as part of rule-based decision-making, understand and execute the specified steps. Regulatory guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, Appendix A, includes several categories of "typical safety-related activities that should be covered by written procedures." The sample should include appropriate procedures in each relevant category:

  - administrative procedures

  - general plant operating procedures

  - procedures for startup, operation, and shutdown of safety-related systems

  - procedures for abnormal, off normal, and alarm conditions

  - procedures for combating emergencies and other significant events

  - procedures for control of radioactivity

  - procedures for control of measuring and test equipment and for surveillance tests, procedures, and calibration

  - procedures for performing maintenance

  - chemistry and radiochemical control procedures

- Range of knowledge-based tasks—these are tasks that are not as well defined by detailed procedures. Knowledge-based decision-making involves greater reasoning about safety and operating goals and the various means of achieving them. A situation may call for knowledge-based decision-making if the rules do not fully address the problem, or the selection of an appropriate rule is not clear. An example in a pressurized water reactor plant may be the difficulty in diagnosing a steam generator tube rupture (SGTR) with a failure of radiation monitors on the secondary side of the plant because: (1) there is no main indication of the rupture (the presence of radiation in secondary side); and (2) the other effects of the rupture (i.e., slight changes in pressures and levels on the primary and secondary sides) may be attributed to other causes. While the operators may use procedures to treat the symptoms of the event, the determination that the cause is an SGTR may warrant situation assessment based on an understanding of the plant's design and the possible combinations of failures that could result in the observed symptoms. Errors in rule-based decision-making result from selecting the wrong rule or incorrectly applying a rule. Errors in knowledge-based decision-making result from mistakes in higher-level cognitive functions such as judgment, planning, and analysis. The latter are more likely to occur in complex failure events where the symptoms do not resemble the typical case, and thus, are not amenable to pre-established rules.

- Range of human cognitive activities—The sample should include the range of cognitive activities performed by personnel, including:

  - detection and monitoring (e.g., of critical safety-function threats)

  - situation assessment (e.g., interpretation of alarms and displays for diagnosis of faults in plant processes and automated control and safety systems)

  - response planning (e.g., evaluating alternatives for recovery from plant failures)

  - response implementation (e.g., in-the-loop control of plant systems, assuming manual control from automatic control systems, and carrying out complicated control actions)

  - obtaining feedback (e.g., of the success of actions taken)

- Range of human interactions—the sample should reflect the range of interactions among plant personnel, including tasks that are performed independently by individual crew members and tasks that are performed by crew members acting as a team. These interactions among plant personnel should include interactions between:

  - MCR operators (e.g., operations, shift turnover walkdowns)

  - MCR operators and auxiliary operators

  - MCR operators and support centers (e.g., the TSC and the emergency offsite facility)

  - MCR operators with plant management, NRC, and other outside organizations

- Tasks that are performed with high frequency.

Evaluation of Criterion (2)

As stated in the evaluation of Criterion (1) above, ISV Plan, Section 5.1.1, and ISV Scenario Plan, Appendix E, list the various evolutions (both high frequency and less common tasks), transients, accidents, and risk-important HAs that are included in the [ ] scenarios. The ISV Scenario Plan, Appendix A, provides the [ ] actual scenarios. This scenario information was compared with the commitments of WCAP-15860 and with the NUREG-0711 criteria. The ISV documents satisfy both documents.

The risk-important HAs and tasks are identified in TR-59/WCAP-16555. WCAP-16555, Section 3.2, identifies [ ] post-accident risk-important HAs in Table 3.2-2. The ISV Plan draft included essentially all of these [ ] risk-important HAs in scenarios. However, the risk-important HA to [ ] was excluded in the ISV scenario Plan. An RAI-SRP18-COLP-53 was written. In the ISV Scenario Plan, this risk-important HA is now included as part of Scenario 18, "Loss of RNS during Mid-Loop Operation." Thus, all risk-important HAs are now addressed in at least one ISV scenario (a few are included in two scenarios). This risk-important HA also has local aspects that cannot be adequately simulated as part of ISV. Thus, the actual verification of acceptability of planned local actions associated with the [ ] will need to be deferred until the plant is built. Therefore, the RAI response proposes adding this to the HFE Design Verification at Plant Startup, APP-OCS-GEH-520. The staff reviewed APP-OCS-GEH-520, Revision B, submitted in a letter dated August 2, 2010, and found that verification of local control action has been added to the document.

The ISV Scenario Plan, Appendix F, previously listed the PRA risk-dominant systems for AP1000. This appendix was deleted from Revision B but is still available in the AP1000 PRA. These systems were verified to all be addressed in the scenarios. [ ].

The following important tasks identified from OSA analyses were included in the ISV, as shown in the ISV Scenario Plan, Appendix E, Table E-1: "Loss of DDS"; [ ]. The following OER important tasks were also identified and included in the ISV Table E-1 and the ISV scenarios: [ ].

The scenarios presented in WCAP-15860 and the ISV Scenario Plan were also found to adequately address a broad range of: procedure–guided tasks, human cognitive activities, and human interactions. Thus, the ISV scenarios were found to adequately address the types of personnel tasks specified in the NUREG-0711 Criterion 2 pending confirmation of RAI-SRP18-COLP-53.

(3) The sample should reflect a range of situational factors that are known to challenge human performance, such as:

- Operationally difficult tasks—The sample should address tasks that have been found to be problematic in the operation of nuclear power plants (e.g., procedure versus situation assessment conflicts). The specific tasks selected should reflect the operating history of the type of plant being validated (or the plant's predecessor).

- Error-forcing contexts—Situations specifically designed to create human errors should be included to assess the error tolerance of the system and the capability of operators to recover from errors should they occur.

- High-workload conditions—The sample should include situations where human performance variation due to high workload and multitasking situations can be assessed.

- Varying-workload situations—The sample should include situations where human performance variation due to workload transitions can be assessed. These include conditions that exhibit: (1) a sudden increase in the number of signals that must be detected and processed following a period in which signals were infrequent; and (2) a rapid reduction in signal detection and processing demands following a period of sustained high task demand.

- Fatigue and circadian factors—The sample should include situations where human performance variation due to personnel fatigue and circadian factors can be assessed.

- Environmental factors—The sample should include situations where human performance variation due to environmental conditions such as poor lighting, extreme temperatures, high noise, and simulated radiological contamination can be assessed.

Evaluation of Criterion (3)

The ISV Plan and ISV Scenario Plan address operationally difficult tasks as identified via the OSA analyses and through the OER. These are summarized in the ISV Scenario Plan, Appendix F, and are discussed in the review of Criterion (2) above. The scenarios have [ ]. This is described in the ISV Plan, Section 5.1.3, "Complications." These complications also are added to the transient or accident scenarios to [ ].

The applicant took exception to addressing fatigue and circadian factors, relying on APP-OCS-GEH-320, Section 5.1.3, which states that the ISV does not address fatigue and circadian factors since it is considered to be impractical to attempt to mimic the conditions that are typical on the operating site. The staff agrees with this position and notes that 10 CFR Part 26, "Fitness for duty programs," Subpart I addresses managing fatigue.

[ ]. For example, [ ]. Other environmental factors are addressed as part of APP-OCS-GEH-520.

Thus, the ISV plans acceptably address Criterion 3.

18.11.8.2.3.2 Identification of Scenarios

The results of the sampling should be combined to identify a set of scenarios to guide subsequent analyses. A given scenario may combine many of the characteristics identified by the operational event sampling.

Evaluation of Criterion (1)

The [ ] scenarios have been developed for use in ISV. The scenarios are quite varied and they do combine the various characteristics outlined in the operational event sampling. Detailed scenario descriptions are provided in the ISV Scenario Plan. The documents reviewed satisfy this criterion.

The scenarios should not be biased in the direction of over representation of the following:

- scenarios for which only positive outcomes can be expected
- scenarios that for integrated system validation are relatively easy to conduct administratively (scenarios that place high demands, data collection or analysis are avoided)
- scenarios that for integrated system validation are familiar and well structured (e.g., which address familiar systems and failure modes that are highly compatible with plant procedures such as “textbook” design-basis accidents)

Evaluation of Criterion (2)

As noted above under Criterion 1, a robust set of [ ] scenarios has been developed and they are described in the ISV Scenario Plan. These scenarios have many failure events both as the key item of the scenario and as peripheral issues. They are not limited to familiar, typical, or easy-to-conduct scenarios or those with only positive outcomes. The documents reviewed satisfy this criterion.

18.11.8.2.4 Integrated System Validation

The objective of reviewing integrated system validation methodology is to verify that the applicant’s methodology will validate the integrated system design (i.e., hardware, software, and personnel elements) using performance-based tests that will determine whether it acceptably supports safe operation of the plant.

18.11.8.2.4.1 Test Objectives

(1) Detailed objectives should be developed to provide evidence that the integrated system adequately supports plant personnel in the safe operation of the plant. The test objectives and scenarios should be developed to address aspects of performance that are affected by the modification [of the] design, including personnel functions and tasks affected by the modification. The objectives should be to:

    - Validate the role of plant personnel.

- Validate that the shift staffing, assignment of tasks to crew members, and crew coordination (both within the control room as well as between the control room and local control stations and support centers) is acceptable. This should include validation of the nominal shift levels, minimal shift levels, and shift turnover.

- Validate that for each human function, the design provides adequate alerting, information, control, and feedback capability for human functions to be performed under normal plant evolutions, transients, design-basis accidents, and selected, risk-significant events that are beyond-design basis.

- Validate that specific personnel tasks can be accomplished within time and performance criteria, with a high degree of operating crew situation awareness, and with acceptable workload levels that provide a balance between a minimum level of vigilance and operator burden. Validate that the operator interfaces minimize operator error and provide for error detection and recovery capability when errors occur.

- Validate that the crew can make effective transitions between the HSIs and procedures in the accomplishment of their tasks and that interface management tasks such as display configuration and navigation are not a distraction or undue burden.

- Validate that the integrated system performance is tolerant of failures of individual HSI features.

- Identify aspects of the integrated system that may negatively affect integrated system performance.

Evaluation of Criterion (1)

The objectives of the AP1000 ISV are identified in Section 4.2 of WCAP-15860, which has been approved by NRC as part of the original AP1000 DC. They included:

1. Establish the adequacy of the integrated HSI for achieving HFE program goals
2. Confirm allocation of function and the structure of tasks assigned to personnel
3. Validate the EOPs and associated HSI
4. Confirm the dynamic aspects of the HSI for task accomplishment
5. Evaluate and demonstrate error tolerance to human and system failures
6. Establish the adequacy of staffing and of the HSI to support staff to accomplish their tasks

These objectives have been included in Section 1.2 of the ISV Plan. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.2 Validation Testbeds

(1) Interface Completeness—The testbed should completely represent the integrated system. This should include HSIs and procedures not specifically required in the test scenarios. For example, adjacent controls and displays may affect the ways in which personnel use those that are addressed by a particular validation scenario.

Evaluation of Criterion (1)

(2) ISV Plan, Section 2, indicates that the ISV will be performed at a dedicated, purpose-built facility. The facility will employ a high fidelity, near full-scope simulator to represent the AP1000 systems and the MCR. This simulator will satisfy the general requirements of Sections 3 and 4 of American National Standards Institute/American National Standard (ANSI/ANS)-3.5-1998, “Nuclear Power Plant Simulators for Use in Operator Training and Examination.”

NUREG-0711 indicates the use of ANSI/ANS-3.5-1998 is an acceptable testbed. This satisfies Criterion 1.

(2) Interface Physical Fidelity—A high degree of physical fidelity in the HSIs and procedures should be represented, including presentation of alarms, displays, controls, job aids, procedures, communications, interface management tools, layout and spatial relationships.

Evaluation of Criterion (2)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1). In addition, in the ISV Plan, Section 5.1.2, “Procedures,” the applicant states that the following types of procedures for AP1000 are incorporated into ISV Scenario Plan, and will be used:

- Optimal Recovery
- Functional Recovery
- Shutdown Procedures
- Normal Operating Procedures (NOPs)
- Abnormal Operating Procedures (AOPs)
- Refueling and Outage Procedures
- Alarm Response Procedures (ARPs)
- Maintenance and Surveillance Guidelines
- [ ]
- [ ]
- [ ]

This approach acceptably meets the staff’s review criterion.

(3) Interface Functional Fidelity—A high degree of functional fidelity in the HSIs and procedures should be represented. All HSI functions should be available. High functional fidelity includes HSI component modes of operation, i.e., the changes in functionality that can be invoked on the basis of personnel selection and/or plant states.

Evaluation of Criterion (3)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(4) Environment Fidelity—A high degree of environment fidelity should be represented. The lighting, noise, temperature, and humidity characteristics should reasonably reflect that

expected. Thus, noise contributed by equipment, such as air handling units and computers should be represented in validation tests.

Evaluation of Criterion (4)

Due to the constraints of the building, the simulator does not include the passive cooling fins; instead, there is a conventional office building tiled ceiling. This results in the lighting system being somewhat different, although it is still representative of the final lighting system design. In addition, the heating and ventilation is provided by a conventional office building system and is, therefore, not representative of the final as-built MCR. Also, the acoustic properties cannot be completely replicated, although they will be similar (i.e., painted walls, hard ceiling tiles). The simulator will be as representative as possible of the final MCR design, so that the design can be assessed. The applicant believes the differences will have minimal or no impact on ISV crew performance. [ ]. Also, the applicant noted in its response to RAI-SRP18-COLP-49, dated February 2, 2010, that the environmental conditions will be fully assessed in APP-OCS-GEH-520. This approach provides sufficient environmental fidelity for ISV and acceptably meets Criterion 4.

(5) Data Completeness Fidelity—Information and data provided to personnel should completely represent the plant systems monitored and controlled from that facility.

Evaluation of Criterion (5)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(6) Data Content Fidelity—A high degree of data content fidelity should be represented. The information and controls presented should be based on an underlying model that accurately reflects the reference plant. The model should provide input to the HSI in a manner such that information accurately matches that which will actually be presented.

Evaluation of Criterion (6)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(7) Data Dynamics Fidelity—A high degree of data dynamics fidelity should be represented. The process model should be capable of providing input to the HSI in a manner such that information flow and control responses occur accurately and in a correct response time; e.g., information should be provided to personnel with the same delays as would occur in the plant.

Evaluation of Criterion (7)

The AP1000 testbed will acceptably meet this criterion (see the evaluation of Criterion 1).

(8) For important actions at complex HSIs remote from the MCR, where timely and precise HAs are required, the use of a simulation or mockup should be considered to verify that human performance requirements can be achieved. (For less risk-important HAs or where the HSIs are not complex, human performance may be assessed based on analysis such as task analysis rather than simulation.)

Evaluation of Criterion (8)

The use of local control stations (LCSs) and the Remote Shutdown Workstation (RSW) are included in the ISV scenarios. Scenario 7 for [ ] is included in the ISV Scenario Plan. The RSW will be validated using [ ]. Operators will be able to [ ]. ISV Plan, Section 2.1, "Physical Scope and Fidelity", describes the details of the simulated RSW panel. Other LCSs are also included in the ISV Scenario Plan. The one local action that is risk important relates to [ ]. The actual verification of acceptability of planned local actions [ ] will need to be deferred until the plant is built. The staff reviewed APP-OCS-GEH-520 against the response to RAI-SRP18-COLP-53 R1 received in a letter dated May 21, 2010. The staff confirmed the document conforms to the RAI response.

(9) The testbeds should be verified for conformance to the testbed characteristics identified above before validations are conducted.

Evaluation of Criterion (9)

Section 2.3 of the ISV Plan describes the simulator testing to be performed prior to ISV evaluations. The ISV Plan references the ISV Scenario Plan for detail concerning how the testing will be performed. That information is provided in ISV Scenario Plan, Appendix C, "Simulator Testing." The objective of this simulator testing in preparation for ISV is to demonstrate that the simulator responds in a manner similar to the reference unit while utilizing the operating procedures and that it meets ANSI/ANS 3.5-1998. The testing will be carried out [ ]. This includes an estimated [ ] of testing. In addition, the ISV Plan, Section 3.3 describes pilot testing of each ISV scenario to ensure simulator readiness, and to confirm effective functioning of test protocols and data collection. This will be done by personnel different from the test subjects.

This provides an acceptable and comprehensive approach to testbed verification. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.3 Plant Personnel

(1) Participants in the validation tests should be representative of actual plant personnel who will interact with the HSI, e.g., licensed operators rather than training or engineering personnel.

Evaluation of Criterion (1)

Section 4.9 of the ISV Plan indicates that validation crews will consist of currently qualified operating crews. The applicant takes partial exception to this item from the program plan, as described in the ISV Plan, Section 1.5, Item 1. Since the AP1000 is a new plant design, the ISV participants will not be fully qualified and experienced AP1000 operators. The ISV subjects will not have the same task performance proficiency as that of fully qualified AP1000 operators. This is reasonable for ISV, and it does make the ISV somewhat more demanding. In accordance with Section 4.1, "Subjects," of the ISV Plan, the ISV subjects will be comprised of the following:

1. A group that has completed the [ ].

2. A group that has partially completed the [ ]. These subjects will have undertaken [ ] training program. Subjects will comprise [ ]. They will have completed [ ] of AP1000 systems training and [ ] of procedure/simulator based training.

3. A limited group of [ ].

Thus, The ISV crews are samples taken from the crews of the AP1000 customer utilities, as described above. This approach acceptably meets the staff's review criterion.

(2) To properly account for human variability, a sample of participants should be used. The sample should reflect the characteristics of the population from which the sample is drawn. Those characteristics that are expected to contribute to system performance variation should be specifically identified and the sampling process should provide reasonable assurance that variation along that dimension is included in the validation. Several factors that should be considered in determining representativeness include: license and qualifications, skill/experience, age, and general demographics.

Evaluation of Criterion (2)

Section 4.1.1 of the ISV Plan discussed participant selection. Two utilities will be providing [ ] crews each. A set of criteria will be provided to the utilities to guide selection of crew members. The criteria include successful completion of training, age range, skills and abilities range, qualifications variation, and prior experience variation. In addition, no operators participating in previous AP1000 tests will be used. This approach acceptably meets the staff's review criterion.

(3) In selection of personnel, consideration should be given to the assembly of minimum and normal crew configurations, including shift supervisors, reactor operators, shift technical advisors, etc., that will participate in the tests.

Evaluation of Criterion (3)

In Section 4.1.2, "Crew Size and Number," of the ISV Plan, the applicant states that the typical crew size for ISV will be [ ]. The crew size will also be varied as a complication in specific scenarios. The maximum control room staff is also specified in Section 4.1.2 and consists of [ ] personnel. The maximum staffing level will be addressed by [ ]. The actual staffing level is specified for each scenario in the ISV Scenario Plan, Section A.n.3, "Scenario Participants" (for n = 1 to [ ]). Scenario participants vary from [ ] to [ ]. This reasonably addresses minimum, normal and maximum staffing levels, plus other values in between, and is acceptable.

In its RAI-SRP18-COLP-26 response of July 12, 2010, the applicant committed to delete TR-52 from the DCD and replace it with APP-OCS-GJR-003, Revision 2, "AP1000 Main Control Room Staff Roles and Responsibilities" to document new staffing values. This is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

(4) To prevent bias in the sample, the following participant characteristics and selection practices should be avoided:

- participants who are [ ]
- participants in [ ]
- participants who are selected for some specific characteristic, such as using crews that are identified as good or experienced.

Evaluation of Criterion (4)

As described in ISV Plan, Section 4.1, “Subjects”; Section 4.1.1, “Selection”; and Section 4.1.2, “Crew Size and Number”, the applicant will use COL utility personnel as test participants and not design personnel. Section 4.1 states that care will be taken to ensure that the test participants do not obtain any prior knowledge of the scenarios to be used in ISV. The participants will not include subjects that participated in the HFE Tests. Section 4.1.1 states that participating utilities will be requested to assign typical crews for ISV testing based on availability. Crews will not be selected for ISV based on individual characteristics. This approach acceptably meets the staff’s review criterion.

18.11.8.2.4.4 Scenario Definition

(1) The operational conditions selected for inclusion in the validation tests should be developed in detail so they can be performed on a simulator. The following information should be defined to provide reasonable assurance that important performance dimensions are addressed and to allow scenarios to be accurately and consistently presented for repeated trials:

- description of the scenario and any pertinent "prior history" necessary for personnel to understand the state of the plant upon scenario start-up

- specific initial conditions (precise definition provided for plant functions, processes, systems, component conditions and performance parameters, e.g., similar to plant shift turnover)

- events (e.g., failures) to occur and their initiating conditions, e.g., time, parameter values, or events

- precise definition of workplace factors, such as environmental conditions

- task support needs (e.g., procedures and technical specifications)

- staffing objectives

- communication requirements with remote personnel (e.g., load dispatcher via telephone)

- the precise specification of what, when and how data are to be collected and stored (including videotaping requirements, questionnaire and rating scale administrations)

- specific criteria for terminating the scenario.

Evaluation of Criterion (1)

The ISV Scenario Plan provides scenario descriptions for each of the [ ] scenarios to be part of ISV. Each scenario in the document includes the following:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

Each scenario provides the above information in acceptable detail. At this time, only three scenarios have complete observer guides. These three provide an acceptable example of how the remaining observer guides will be completed. This approach acceptably meets the staff's review criterion.

(2) Scenarios should have appropriate task fidelity so that realistic task performance will be observed in the tests and so that test results can be generalized to actual operation of the real plant.

Evaluation of Criterion (2)

This criterion is addressed through the use of a simulation facility for ISV that satisfies the general requirements of Sections 3 and 4 of ANSI/ANS-3.5-1988; use of COL plant operating personnel in training for operations; and use of realistic but challenging scenarios. This approach conforms to NUREG-0711 Criterion 2.

(3) When evaluating performance associated with operations remote from the MCR, the effects on crew performance due to potentially harsh environments (i.e., high radiation) should be realistically simulated (i.e., additional time to don protective clothing and access radiologically controlled areas).

Evaluation of Criterion (3)

The ISV Plan notes that the use of LCSs is included in the ISV scenarios and they will use simulated interactions with local operations that extend beyond the MCR. Scripted responses will be provided for the operations support staff to perform specified roles as plant personnel in applicable scenarios (e.g., local operators). This is acceptable.

There is one local control action that is a risk-important HA. This is [ ] and is included in an ISV scenario. The actual verification of acceptability of planned local actions associated with the hatches will need to be deferred until the plant is built. This is an acceptable approach. Therefore, the RAI response proposes adding this to APP-OCS-GEH-520. The staff reviewed APP-OCS-GEH-520, Revision B, submitted in a letter dated August 2, 2010, and found that verification of local control action has been added to the document.

18.11.8.2.4.5 Performance Measurement

The review of performance measurement covers measurement characteristics, performance measure selection, and performance criteria.

18.11.8.2.4.5.1 Measurement Characteristics

(1) Performance Measurement Characteristics—Performance measures should acceptably exhibit the following measurement characteristics to provide reasonable assurance that the measures are of good quality (it should be noted that some of the characteristics identified below may not apply to every performance measure):

- Construct Validity—A measure should accurately represent the aspect of performance to be measured.

- Diagnosticity—A measure should provide information that can be used to identify the cause of acceptable or unacceptable performance.

- Impartiality—A measure should be equally capable of reflecting good as well as bad performance.

- Objectivity—A measure should be based on phenomena that are easily observed.

- Reliability—A measure should be repeatable; i.e., if the same behavior is measured in exactly the same way under identical circumstances, the same measurement result should be obtained.

- Resolution—A measure should reflect the performance at an appropriate level of resolution, i.e., with sufficient detail to permit a meaningful analysis.

- Sensitivity—A measure's range (scale) and the frequency of measurement (how often data are collected) should be appropriate to the aspect of performance being assessed.

- Simplicity—A measure should be simple both from the standpoint of executing the tests and from the standpoint of communicating and comprehending the meaning of the measures.

- Unintrusiveness—A measure should not significantly alter the psychological or physical processes that are being investigated.

Evaluation of Criterion (1)

Section 6 of the ISV Plan describes the performance measures to be used to evaluate integrated system performance. The characteristics of the measures are addressed in Section 6.2. Several of the measures are well-known, commonly used measures with established, acceptable measurement characteristics, such as the [ ]. For others developed by Westinghouse, the basis of the measures is identified. For example, a measure of team performance will be used that is based on a [ ]. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.5.2 Performance Measure Selection

(1) A hierarchal set of performance measures should be used, which includes measures of the performance of the plant and personnel (i.e., personnel tasks, situation awareness, cognitive workload, and anthropometric/physiological factors). Some of these measures could be used as "pass/fail" criteria for validation and the others to better understand personnel performance and to facilitate the analysis of performance errors. The applicant should identify which are in each category.

Evaluation of Criterion (1)

Section 6.1 of the ISV Plan describes the measures to be used. The measures are hierarchal including [ ]. Thus, an acceptable hierarchal set of performance measures will be used to assess integrated system performance.

Section 6.3.1 of the ISV Plan identifies the measures to be used as pass/fail (P/F) criteria. P/F measures are measures reflecting [ ]. This provides a reasonable set of measures to serve as P/F criteria.

Performance measures to be used as diagnostic measures are discussed in ISV Plan, Section 6.3.2. The measures are listed in Table 6.3-2 and include all measures collected during ISV trials with the exception of the P/F measures.

This approach acceptably meets the staff's review criterion.

(2) Plant Performance Measurement—Plant performance measures representing functions, systems, components, and HSI use should be obtained.

Evaluation of Criterion (2)

Section 6 of the ISV Plan discusses ISV performance measurement. P/F plant-level measures involve applicable technical [ ]. Plant-level diagnostic measures are also defined for each scenario so that [ ]. For example, for the [ ]. This approach acceptably meets the staff's review criterion.

(3) Personnel Task Measurement—For each specific scenario, the tasks that personnel are [needed] to perform should be identified and assessed. Two types of personnel tasks should be measured: primary (e.g., start a pump), and secondary (e.g., access the pump status display). Primary tasks are those involved in performing the functional role of the operator to supervise the plant; i.e., monitoring, detection, situation assessment,

response planning, and response implementation. Secondary tasks are those personnel [need to] perform when interfacing with the plant, but which are not directed to the primary task, such as navigation and HSI configuration. This analysis should be used for the identification of potential errors of omission.

- Primary tasks should be assessed at a level of detail appropriate to the task demands. For example, for some simple scenarios, measuring the time to complete a task may be sufficient. For more complicated tasks, especially those that may be described as knowledge-based, it may be appropriate to perform a more fine-grained analysis such as identifying task components: seeking specific data, making decisions, taking actions, and obtaining feedback. Tasks that are important to successful integrated system performance and are knowledge-based should be measured in a more fine-grained approach.

- The measurement of secondary tasks should reflect the demands of the detailed HSI implementation, e.g., time to configure a workstation, navigate between displays, and manipulate displays (e.g., changing display type and setting scale).

- The tasks that are actually performed by personnel during simulated scenarios should be identified and quantified. (Note that the actual tasks may be somewhat different from those that should be performed). Analysis of tasks performed should be used for the identification of errors of commission.

- The measures used to quantify tasks should be chosen to reflect the important aspects of the task with respect to system performance, such as:

    - time
    - accuracy
    - frequency
    - errors (omission and commission)
    - amount achieved or accomplished
    - consumption or quantity used
    - subjective reports of participants
    - behavior categorization by observers

Evaluation of Criterion (3)

Section 6 of the ISV Plan discusses ISV performance measurement. Measurement of operator tasks involves both P/F and diagnostic variables. Successful performance of risk-important HAs is a P/F variable. For example, for the [ ] as tasks to assess using P/F measures. For diagnostic purposes, the performance of key tasks is measured. These actions are listed in the observer guides for each scenario. This approach acceptably meets the staff's review criterion.

(4) Situation Awareness—Personnel situation awareness should be assessed. The approach to situation awareness measurement should reflect the current state-of-the-art.

Evaluation of Criterion (4)

Section 6 of the ISV Plan discusses ISV performance measurement. Situation awareness is measured using the [ ]. SART is a widely used and acceptable measure of situation awareness. This approach acceptably meets the staff's review criterion.

(5) Cognitive Workload—Personnel workload should be assessed. The approach to workload measurement should reflect the current state-of-the-art.

Evaluation of Criterion (5)

Section 6 of the ISV Plan discusses ISV performance measurement. Cognitive workload is measured using the [ ]. This approach acceptably meets the staff's review criterion.

(6) Anthropometric and Physiological Factors— Anthropometric and physiological factors include such concerns as visibility of indications, accessibility of control devices, and ease of control device manipulation that should be measured where appropriate. Attention should be focused on those aspects of the design that can only be addressed during testing of the integrated system, e.g., the ability of personnel to effectively use the various controls, displays, workstations, or consoles in an integrated manner.

Evaluation of Criterion (6)

Section 6 of the ISV Plan discusses ISV performance measurement. Information on general aspects of anthropometrics, including control room layout and workstation configuration, have been included in the [ ]. An assessment of anthropometric and physiological factors will also be made during HFE Design Verification. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.5.3 Performance Criteria

(1) Criteria should be established for the performance measures used in the evaluations. The specific criteria that are used for decisions as to whether the design is validated or not should be specified and distinguished from those being used to better understand the results.

Evaluation of Criterion (1)

Section 6.3, "Criteria," of the ISV Plan discusses the criteria to be used in evaluating performance measures. P/F performance measures are used to validate the design as was discussed in Section 4.5.2, "Performance Measure Selection, of this report. The general acceptance criteria are (1) [ ].

For diagnostic measures, criteria are identified in ISV Plan, Table 6.3-2. The table provides criteria for all diagnostic measures. For example, the criteria for evaluating workload include: (1) average rating of workload across subjects from questionnaire is < 85 (range 0 to 100); (2) subjects demonstrate behavior, as specified in the scenario description for each scenario, that their workload is within a reasonable range and there are no indications of stress caused by excessive workload; and (3) No workload issues are identified through questionnaire comments, debriefing, video and audio recording review. Failure to meet the criteria is evaluated by the HED resolution process (APP-OCS-GEH-420) to determine its priority.

This approach acceptably meets the staff's review criterion.

(2) The basis for criteria should be defined, e.g., requirement-referenced, benchmark referenced, normative referenced, and expert-judgment referenced.

Evaluation of Criterion (2)

Section 6.3, "Criteria," of the ISV Plan discusses the criteria to be used in evaluating performance measures. The basis for criteria for P/F measures is [ ]. The criteria established for diagnostic measures are based [ ]. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6 Test Design

18.11.8.2.4.6.1 Coupling Crews and Scenarios

(1) Scenario Assignment—Important characteristics of scenarios should be balanced across crews. Random assignment of scenarios to crews is not recommended. The value of using random assignment to control bias is only effective when the number of crews is quite large. Instead, the validation team should attempt to provide each crew with a similar and representative range of scenarios.

Evaluation of Criterion (1)

Section 3.2 of the ISV Plan discussed assignment of participants to trials. A final run order will be identified after pilot testing so aspects such as scenario duration can be determined. However, an example is provided in Table 3.3-1 of a counter balanced presentation of scenarios to crews. In the example, [ ]. Assignments are made [ ]. The constraints and considerations are clearly identified in the ISV plan. This approach acceptably meets the staff's review criterion.

(2) Scenario Sequencing—The order of presentation of scenario types to crews should be carefully balanced to provide reasonable assurance that the same types of scenarios are not always being presented in the same linear position, e.g., the easy scenarios are not always presented first.

Evaluation of Criterion (2)

Section 3.2 of the ISV Plan discussed assignment of participants to trials. As noted in the evaluation of Criterion (1), the final trial orders will be determined following pilot testing. One of the principles to be followed in that determining the final run order is to balance the order to accommodate the types of concerns raised in the review criterion. For example, the Plan states that "[ ]." Such considerations should minimize the possibility of linear position effects. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.2 Test Procedures

(1) Detailed, clear, and objective procedures should be available to govern the conduct of the tests. These procedures should include:

- The identification of which crews receive which scenarios and the order that the scenarios should be presented.

- Detailed and standardized instructions for briefing the participants. The type of instructions given to participants can affect their performance on a task. This source of bias can be minimized by developing standard instructions.

- Specific criteria for the conduct of specific scenarios, such as when to start and stop scenarios, when events such as faults are introduced, and other information discussed in Section 11.4.3.2.4, "Scenario Definition."

- Scripted responses for test personnel who will be acting as plant personnel during test scenarios. To the greatest extent possible, responses to communications from operator participants to test personnel (serving as surrogate for personnel outside the control room personnel) should be prepared. There are limits to the ability to preplan communications since personnel may ask questions or make requests that were not anticipated. However, efforts should be made to detail what information personnel outside the control room can provide, and script the responses to likely questions.

- Guidance on when and how to interact with participants when simulator or testing difficulties occur. Even when a high-fidelity simulator is used, the participants may encounter artifacts of the test environment that detract from the performance for tasks that are the focus of the evaluation. Guidance should be available to the test conductors to help resolve such conditions.

- Instructions regarding when and how to collect and store data. These instructions should identify which data are to be recorded by:

  - simulation computers

  - special purpose data collection devices (such as situation awareness data collection, workload measurement, or physiological measures)

  - video recorders (locations and views)

  - test personnel (such as observation checklists)

  - subjective rating scales and questionnaires.

- Procedures for documentation, i.e., identifying and maintaining test record files including crew and scenario details, data collected, and test conductor logs. These instructions should detail the types of information that should be logged (e.g., when tests were performed, deviations from test procedures, and any unusual events that may be of importance to understanding how a test was run or interpreting test results) and when it should be recorded.

Evaluation of Criterion (1)

The evaluation below is numbered to correspond to the bulleted criteria above.

1. Section 3.2 of the ISV Plan addresses crew assignment to scenarios. See the discussion of crew assignments in Section 4.6.1, "Coupling Crews and Scenario," Criterion 1 of this report above. This acceptably meets the subcriterion.

2. Section 5.2 of the ISV Plan addresses the requirements for crew briefing in general. The briefing will [ ]. Detailed information on crew briefings is included in the scenario descriptions for Scenarios 1, 2 and 12 in the ISV Scenario Plan. The information provided in these three detailed scenarios is complete and consistent with the high-level guidance in the ISV Plan. For example, the briefing for Scenario 2 is:

   [ ].

   This acceptably meets the subcriterion.

3. [ ]:

   - [ ]
   - [ ]
   - [ ]

   This acceptably meets the subcriterion.

4. Section 5.2.2, "Communications with ISV Personnel," of the ISV Plan describes the general approach for communicating with ISV crews. The Plan indicates that scripted responses will be used when test personnel act as plant personnel, such as a local operator. The Plan further states that "[ ]." These scripted responses are included in the ISV Scenario Plan. For example, for Scenario 1, the following instruction is provided:

   [ ].

   This acceptably meets the subcriterion.

5. Section 5.2.3, "Unforeseen Events," of the ISV Plan provides guidance on interacting with participants when unexpected difficulties arise. The guidance addresses events unrelated to the testing, such as fire drills, as well as related events, such as simulator anomalies. The plan outlines responsibilities for interacting with crews and guidance on resuming versus restarting trials. This acceptably meets the subcriterion.

6. Section 5.2.1, "General Procedures and Documentation," and Section 5.2.4, "Storage of Data," of the ISV Plan define the responsibilities and procedures for management of ISV

data. For example, the Plan identifies the ISV coordinator as the individual responsible for data management. With regard to simulator recorded data, the Plan indicates that "The discrete event data and plant parameter data from the simulator will be stored on a server and burnt onto discs. The file names for this data will identify the scenario number, the crew, and will be dated and time-stamped."

Further, the ISV Plan indicates that "At the end of each scenario, the ISV Coordinator will distribute and collect the completed post-trial questionnaires for the subjects, and at the end of the crews and observers participation in ISV, the ISV Coordinator will distribute and collect the final questionnaires for the subjects and observers. All of this information is hardcopy, and will be clearly marked and stored in a secure location."

This acceptably meets the subcriterion.

7. Section 5.2, "ISV Procedures," of the ISV Plan provides procedures for documenting all data collected during the ISV. The guidance in the ISV Plan addresses all forms of data, e.g., simulator logs and questionnaires. The procedures are sufficiently explicit to ensure data is not mishandled or lost (see examples in the evaluations above). This acceptably meets the subcriterion.

In summary, the ISV Plan provides detailed, clear, and objective procedures to govern the conduct of the ISV tests. This approach acceptably meets the staff's review criterion.

(2) Where possible, test procedures should minimize the opportunity of tester expectancy bias or participant response bias.

Evaluation of Criterion (2)

The ISV Plan indicates that observers will be independent of the project and that their assignment to trials will be systematically varied. The use of standardized and scripted responses should also help to minimize bias. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.3 Test Personnel Training

(1) Test administration personnel should receive training on:

- the use and importance of test procedures
- experimenter bias and the types of errors that may be introduced into test data through the failure of test conductors to accurately follow test procedures or interact properly with participants
- the importance of accurately documenting problems that arise in the course of testing, even if due to test conductor oversight or error.

Evaluation of Criterion (1)

Section 4.3 of the ISV Plan addresses training of test personnel. It states that the training for the ISV staff will be sufficient to ensure effective execution of the test scenarios and data collection. This training will occur during the pilot testing of the simulator and ISV scenarios.

The training will be specific for the tasks to be performed during ISV. Test conductor roles will be rehearsed prior to ISV. The training will include how and when to communicate with the participants. Scripted responses will be provided for the operations support staff to perform specified roles as plant personnel in applicable scenarios (e.g., local operators). In addition, training will be given on the importance of the ISV procedures and the possible impact of not following the ISV procedures. This will help ensure consistency of the ISV staff performance and behavior across the scenarios. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.4 Participant Training

(1) Participant training should be of high fidelity; i.e., highly similar to that which plant personnel will receive in an actual plant. The participants should be trained to provide reasonable assurance that their knowledge of plant design, plant operations, and use of the HSIs and procedures is representative of experienced plant personnel. Participants should not be trained specifically to perform the validation scenarios.

Evaluation of Criterion (1)

Sections 4.1 and 4.1.3 of the ISV Plan discuss participant training. Operations personnel from the customer utilities participating in the AP1000 training program will be used for ISV. Training will include both classroom and hands-on simulator components. The training will be developed and delivered by the Westinghouse Training Group.

The training program will provide personnel with detailed AP1000 systems and plant knowledge. The program will be presented using a combination of classroom instruction, self-study, procedure walk through, and exercises. Training will include [ ] of AP1000 systems training and [ ] of procedure/simulator based training. This will include EOPs, AOPs, and GOPs.

This approach acceptably meets the staff's review criterion.

(2) Participants should be trained to near asymptotic performance (i.e., stable, not significantly changing from trial to trial) and tested prior to conducting actual validation trials. Performance criteria should be similar to that which will be applied to actual plant personnel.

Evaluation of Criterion (2)

Sections 4.1 and 4.1.3 of the ISV Plan describe the training program for ISV participants. As discussed under Criterion (1), the program provides sufficient training such that the skill and knowledge levels of participants should not be significantly changing between trials. This approach acceptably meets the staff's review criterion.

18.11.8.2.4.6.5 Pilot Testing

(1) A pilot study should be conducted prior to conducting the integrated validation tests to provide an opportunity to assess the adequacy of the test design, performance measures, and data collection methods.

Evaluation of Criterion (1)

Section 3.3 of the ISV Plan indicates pilot testing will be performed to address these aspects of the test. The objectives of simulator pilot testing are to demonstrate that the simulator responds in a manner similar to the reference unit while utilizing the plant operating procedures, to ensure simulator readiness, and to minimize the likelihood of test failures or delays. The pilot testing will be carried out in [ ], as described in the ISV Scenario Plan, Appendix C. In addition to the testing of the simulator model, thorough pilot testing of all scenarios will be carried out. This approach acceptably meets the staff's review criterion.

(2) If possible, participants who will operate the integrated system in the validation tests should not be used in the pilot study. If the pilot study must be conducted using the validation test participants, then:

- the scenarios used for the pilot study should be different from those used in the validation tests; and

- care should be given to provide reasonable assurance that the participants do not become so familiar with the data collection process that it may result in response bias.

Evaluation of Criterion (2)

Section 3.3 of the ISV Plan states that the participants for ISV will not be involved in pilot testing. Rather, the pilot testing will be performed by the Westinghouse simulator development staff, with support as needed from other Westinghouse personnel. This acceptably addresses Criterion (2). This approach acceptably meets the staff's review criterion.

18.11.8.2.4.7 Data Analysis and Interpretation

(1) Validation test data should be analyzed through a combination of quantitative and qualitative methods. The relationship between observed performance data and the established performance criteria should be clearly established and justified based upon the analyses performed.

Evaluation of Criterion (1)

Section 7, "Processing of Results," of the ISV Plan [ ]. This approach acceptably meets the staff's review criterion.

(2) For performance measures used as pass/fail indicators, failed indicators must be resolved before the design can be validated. Where performance does not meet criteria for the other performance measures, the results should be evaluated using the HED evaluation process.

Evaluation of Criterion (2)

Section 7.3 of the ISV Plan indicates that each scenario is run [ ] times with [ ] different crews. If a scenario fails a P/F criterion an HED is defined and resolved. The scenario is then rerun a minimum of [ ] times with [ ]. With respect to diagnostic measures, observation of a small number of HEDs will result in a [ ] trial being run using a [ ]. This approach will help confirm whether an HED exists or not. The applicant indicated that, if the results of the [ ] trial confirm an issue, an HED will be identified and resolved. The scenario then will be re-run [ ] times using [ ]. This approach acceptably meets the staff's review criterion.

(3) The degree of convergent validity should be evaluated, i.e., the convergence or consistency of the measures of performance.

Evaluation of Criterion (3)

Section 7.2 of the ISV Plan indicates that the degree of convergence of measures will be assessed in the interpretation of the results. The Plan states:

> The degree to which convergent (i.e., consistent) results are observed from different measurement techniques will be [ ] and the results will presented in the ISV results report. The analysis will determine if the different measurement techniques indicate the same problems. If so, it strengthens the conclusion that a problem exists and it needs to be addressed. Likewise, if none of the measurement techniques indicates that there is a problem (i.e., different measurement techniques record successful performance), then it increases the degree of certainty that a problem does not exist.

This approach acceptably meets the staff's review criterion.

(4) The data analyses should be independently verified for correctness of analysis.

Evaluation of Criterion (4)

Section 7.2 of the ISV Plan indicates that independent verification of results will be performed using applicable Westinghouse quality assurance procedures provided in APP-GW-GAP-100, "Inter-Business Unit Edition Policies & Procedures." This approach acceptably meets the staff's review criterion.

(5) The inference from observed performance to estimated real-world performance should allow for margin of error; i.e., some allowance should be made to reflect the fact that actual performance may be slightly more variable than observed validation test performance.

Evaluation of Criterion (5)

Section 6.3.1, "Pass/Fail Criteria," of the ISV Plan indicates that for P/F criteria, each scenario specifies that [ ]. In order to ensure margin, there are also typically acceptance criteria relating to not exceeding the [ ].

Regarding the risk-important HAs, [ ]. Section 6.3.1 of the ISV Plan states that "In a number of cases in the PRA, the [ ]. Therefore, the [ ] to perform the risk-important human actions will be closely monitored. If a case occurs where the [ ] is potentially insufficient to ensure reliable operator performance, this will be identified as [ ]." The method of identifying and documenting the [ ] for performing the risk-important HAs is described in the ISV Scenario Plan, in the Scenario Specifications and in the Observer Guides for the scenarios.

RAI-SRP18-COLP46 requested more information on the mechanism for specifying and documenting the [ ] needed to accomplish the risk-important HAs in the ISV Scenarios. In its response, dated August 2, 2010, the applicant stated that the “[ ]. The HFE analyst will use [ ] in Section 6.2, “Methods,” of the ISV Plan to complete this calculation. This approach limits the potential of results being influenced by the observer’s judgment or the observer missing a task step, event or operator action. The [ ] provide an objective confirmation of the observation results.

This approach acceptably meets the staff’s review criterion.

18.11.8.2.4.8 Validation Conclusions

(1) The statistical and logical bases for determining that performance of the integrated system is and will be acceptable should be clearly documented.

Evaluation of Criterion (1)

Section 7.4 of the ISV Plan provides the commitment to document the basis for validation conclusions. The ISV Plan states “The basis for concluding that the AP1000 MCR, HSI resources, procedures, and operator training are adequate (or not) will be described (i.e., that the integrated system performed acceptably during testing and can be expected to support safe operation in actual use).” This approach acceptably meets the staff’s review criterion.

(2) Validation limitations should be considered in terms of identifying their possible effects on validation conclusions and impact on design implementation. These include:

- aspects of the tests that were not well controlled
- potential differences between the test situation and actual operations, such as absence of productivity-safety conflicts
- potential differences between the validated design and plant as built (if validation is directed to an actual plant under construction where such information is available or a new design using validation results of a predecessor).

Evaluation of Criterion (2)

Section 7.4 of the ISV Plan provides the commitment to document test limitations. This approach acceptably meets the staff’s review criterion.

18.11.8.3 Conclusion

The staff concludes that APP-OCS-GEH-320 and APP-OCS-GEH-321 provide implementation plans that conform to the NUREG-0711 criteria associated with ISV. The staff’s review of the AP1000 ISV Plan and the ISV Scenario Plan concludes that the plans are comprehensive and thorough and provide reasonable assurance that the ISV will effectively identify any operator challenges associated with the HSI design. Open Item OI-SRP18-COLP-03A was created to track the completion of these documents. Based on the preceding information, the staff concludes that this open item and the corresponding element of ITAAC Design Commitment 4 (DCD, Revision 15) are complete and closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.11.9 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 4 of 5, Issue Resolution Verification

18.11.9.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 4): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the …Issue Resolution Verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.9.2 Evaluation

NUREG-0711, Section 11.4.4.2, Criterion 1, states the following:

> Discrepancies could be acceptable within the context of the fully integrated design. If sufficient justification exists, a deviation from the guidelines may not constitute an HED. The technical basis for such a determination could include an analysis of recent literature or current practices, tradeoff studies, or design engineering evaluations and data. The applicant should identify unjustified discrepancies as HEDs to be addressed by the HED resolution.

Evaluation of Criterion 1

Each of the three V&V implementation plans previously referenced (APP-OCS-GEH-120, APP-OCS-GEH-220, and APP-OCS-GEH-320) include sections directing that all discrepancies be documented on a discrepancy worksheet. The Operation and Control System product manager screens the worksheet. If the discrepancy can be directly justified, it is not considered an HED. Unjustified discrepancies are identified as HEDs, and the applicant must address them using the formal resolution process in APP-OCS-GEH-420. The staff concludes that procedures referenced in this section provide sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for HED justification.

NUREG-0711, Section 11.4.4.2, Criterion 2, states the following:

The HED analysis should include the following:

- Plant system—The potential effects of all HEDs relevant to a single-plant system should be evaluated. The potential effects of these HEDs on plant safety and personnel performance should be determined, in part, by the safety significance of the plant system, their effect on the accident analyses summarized in the SAR, and their relationship to risk-significant sequences in the plant PRA.

- HED scope—The scope of the HED should consider the following:

  - Global features HEDs—These HEDs relate to configurational and environmental aspects of the design, such as lighting, ventilation, and traffic flow. They relate to general human performance issues.

  - Standardized features HEDs—These HEDs relate to design features that are governed by the applicant's design guidelines used across various controls and displays of the HSI (e.g., display screen organization and conventions for format, coding, and labeling). Because a single guideline may be used across many aspects of the design, a single HED could be applicable to many personnel tasks and plant systems.

  - Detailed features HEDs—These HEDs relate to design features that are not standardized, thus their generality has to be assessed.

  - Other—This subcategory specifically pertains to HEDs identified from integrated system validation that cannot be easily assigned to any of the three preceding categories.

- Individual HSI or procedure—HEDs should be analyzed with respect to individual HSIs and procedures. The potential effects of these HEDs on plant safety and personnel performance are determined, in part, by the safety significance of the plant system that is related to the particular component.

- Personnel function—HEDs should be analyzed with respect to individual personnel functions. The potential effects of these HEDs is determined, in part, by the importance of the personnel function to plant safety (e.g., consequences of failure) and the cumulative effect on personnel performance (e.g., degree of impairment and types of potential errors).

  The applicant should also analyze HEDs with respect to the cumulative effects of multiple HEDs on plant safety and personnel performance.

  In addition to addressing the specific HEDs, the analysis should treat the HEDs as indications of potentially broader problems.

Evaluation of Criterion 2

In APP-OCS-GEH-420, Section 2.2, the applicant stated that the [ ], which the OCS product manager approves. The assessment [ ]. This prioritization addresses the first bullet from the NUREG-0711 criterion above:

- Priority 1: [ ]
  - [ ].
  - [ ].
- Priority 2: [ ]:
  - [ ]
  - [ ]
  - [ ]
- Priority 3: All others

APP-OCS-GEH-420, Section 2.3, addresses the HED scope. In this section, the applicant stated that the cumulative effects of Priority 1 and 2 HEDs are analyzed by organizing HEDs into the following categories:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

This last category identifies [ ], and others using the same definitions as the NUREG criterion. By separating the [ ]. This addresses the remaining parts of the NUREG-0711 criterion above.

APP-OCS-GEH-420, Section 2.5, states that an [ ]; an evaluation is performed to [ ]. The categorization described above is used to identify these generic implications.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for HED analysis.

NUREG-0711, Section 11.4.4.2, Criterion 3, states the following:

> The applicant should use a systematic evaluation to identify HEDs for correction. Priority 1 HEDs are those with direct, indirect, or potential safety consequences. Priority 2 HEDs are those that do not have significant safety consequences, but do have potential consequences to plant performance/operability, nonsafety-related personnel performance/efficiency, or other factors affecting overall plant operability. The remaining HEDs are those that do not satisfy the criteria associated with the first and second priorities. Resolution of these HEDs

> is not an NRC safety concern but may be resolved at the discretion of the applicant.

Evaluation of Criterion 3

This criterion is addressed in the previous section.

NUREG-0711, Section 11.4.4.2, Criterion 4, states the following:

> The applicant should fully document each HED, including assessment category (priority for correction), associated plant system, associated personnel function, and associated HSI or procedure. The documentation should clearly show whether the HED was dismissed or identified as needing design modification, and the basis for this determination in terms of consequence to plant safety or operation should be clearly described.

Evaluation of Criterion 4

Section 2 of APP-OCS-GEH-420 includes documentation requirements. In summary, when the HFE engineer justifies an HED, he or she [ ]. When an HED is resolved by a design solution, a [ ] is used in conjunction with a [ ] to identify the best solution. Solutions will be consistent with the system requirements used to design the system. Design solutions follow the design process which documents the impact on safety.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for HED evaluation documentation.

NUREG-0711, Section 11.4.4.2, Criterion 5, states the following:

> The applicant should identify design solutions to correct HEDs. The design solutions should be consistent with system and personnel requirements identified in the preparatory analysis (i.e., operating experience review, function and task analysis, and HSI characterization).

Evaluation of Criterion 5

APP-OCS-GEH-420, Section 2.6, states that the design solution will be consistent with the system requirements used to design the system. By comparing proposed changes to the HFE design to the original system requirements, the applicant will ensure the original design basis is maintained or adjusted as necessary.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for development of HED design solutions.

NUREG-0711, Section 11.4.4.2, Criterion 6, states the following:

> The applicant should evaluate designs by repeating the appropriate V&V analyses. When the problems identified by an HED cannot be fully corrected, the applicant should provide appropriate justification.

Evaluation of Criterion 6

In APP-OCS-GEH-420, Section 2.8, the applicant stated that, for design solutions associated with the HFE design verification plan or HFE task support verification plan, independent verifiers will evaluate the HSI design changes using the same standards, guidance, and methodology as described in the applicable verification plan. For design solutions associated with the integrated system validation plan, the human factors team will determine the appropriate evaluation process, using a graded approach, based on the complexity and impact of the design changes. Independent verifiers will then perform the evaluation process.

The staff concludes that APP-OCS-GEH-420 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion for design solution evaluation.

18.11.9.3 Conclusion

The staff concludes that APP-OCS-GEH-420 provides an implementation plan that satisfactorily addresses the NUREG-0711 criteria associated with tracking and resolving HEDs. This implementation plan provides reasonable assurance that issues will be identified during all stages of the design process and that these issues will be prioritized and resolved in an efficient manner. This element of ITAAC Design Commitment 4 (DCD Revision 15) as described above is complete and closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.11.10 Evaluation of Tier 1 Information—Design Commitment 4, ITAAC Table 3.2-1, Tier 1, Section 3.2 (DCD Revision 15), Part 5 of 5, Plant HFE/HSI (as Designed at the Time of Plant Startup) Verification

18.11.10.1 Summary of Technical Information

ITAAC Design Commitment 4 reads as follows:

> Design Commitment: An HFE program verification and validation implementation plan is develop[ed] in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.
>
> Inspection, Test, and Analysis: An inspection of the HFE verification and validation implementation plan will be performed.
>
> Acceptance criteria (part 5): A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan and includes the…. Plant HFE/HSI (as designed at the time of plant startup) Verification activity.

In DCD Revision 17, the applicant deleted this ITAAC based on completion of the work it described.

18.11.10.2 Evaluation

The applicant submitted APP-OCS-GEH-520 to address this part of ITAAC 4. Open Item OI-SRP18-COLP-4A was created to track completion of the staff's review of this document. The acceptance criteria for this implementation plan are found in NUREG-0711 Section 12, "Design Implementation." The staff evaluation of this plan is being provided in this section of the SER so that material applicable to ITAAC 4 closure is kept together.

NUREG-0711, Section 12.4.6, Criterion 1, states the following:

> Aspects of the design that were not addressed in V&V should be evaluated using an appropriate V&V method. Aspects of the design addressed by this criterion may include design characteristics such as new or modified displays for plant-specific design features and features that cannot be evaluated in a simulator such as Control Room lighting and noise.

Evaluation of Criterion 1

In APP-OCS-GEH-520, Section 1, the applicant states that specific aspects of the OCS. HSI design that cannot be evaluated in a simulator will be evaluated via a walk down of the applicable plant area after construction. This plan applies to all control areas included in the HFE scope including the MCR, the remote shutdown room, TSC, radioactive waste control, and LCSs. APP-OCS-GER 120 and APP-OCS-GER-220 document the results of the design and task analysis verification and are used as the basis for identifying design verifications that have not been completed. The procedure specifically identifies lighting, noise, ambient temperature and humidity, the closed circuit TV system, communication facilities, and maintainability as areas that will be evaluated. The applicant indicates that where appropriate physical measurements will be taken for key environmental features including lighting, thermal conditions, and acoustics.

The staff concludes that APP-OCS-GEH-520 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion.

NUREG-0711, Section 12.4.6, Criterion 2, states the following:

> The final (as-built in the plant) HSIs, procedures, and training should be compared with the detailed design description to verify that they conform to the design that resulted from the HFE design process and V&V activities. Any identified discrepancies should be corrected or justified.

Evaluation of Criterion 2

In APP-OCS-GEH-520, the applicant states that the as-built HSIs will be verified to the same as those that resulted from the HFE program. A team is used to complete the verification. The team uses the expected design configuration and control information, including the style guide, detailed design descriptions, and guidance on evaluating maintainability, to compare the as-built design against. The adequacy of procedures and training are addressed as part of the staff's operating program inspections.

The staff concludes that APP-OCS-GEH-520 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion.

NUREG-0711, Section 12.4.6, Criterion 3, states the following:

> All HFE-related issues documented in the issue tracking system should be verified as adequately addressed.

Evaluation of Criterion 3

In APP-OCS-GEH-520, the applicant states that all HEDs will be verified as being adequately addressed.

The staff concludes that APP-OCS-GEH-520 provides sufficient details to satisfactorily demonstrate implementation of this NUREG criterion.

18.11.10.3 Conclusion

The staff concludes that APP-OCS-GEH-520 provides an implementation plan that satisfactorily addresses the NUREG-0711 criteria associated with the as-built design verification. The scope and methods described provide reasonable assurance that the as-built HFE design configuration will mirror the design described in the DC. This element of ITAAC Design Commitment 4 (DCD Revision 15) as described above is complete and closed; therefore, Open Item OI-SRP18-COLP-04A is closed.

The DCD changes provide detailed human factors design information that would otherwise have to be addressed through verification of the ITAAC. Therefore, the changes to the DCD eliminate the need for design acceptance criteria in accordance with the finality criteria in 10 CFR Part 52.63(a)(1)(iv).

18.16 Tier 2* Information

The staff has determined that the following information referenced in DCD Tier 2, Chapter 18, Revision 17, must be designated as Tier 2* information in the AP1000 DCD. This information is in addition to the information identified as Tier 2* for Revision 15 of the DCD as documented in Section 18.16 of NUREG-1793.

1. APP-OCS-GEH-120, Revision B, “AP1000 Human Factors Engineering Design Verification Plan.” (This report explains the applicant’s method for design verification.)

2. APP-OCS-GEH-220, Revision B, “AP1000 Human Factors Engineering Task Support Verification Plan.” (This report explains the applicant’s method for task support verification.)

3. APP-OCS-GEH-320, Revision D, “AP1000 Human Factors Engineering Integrated System Validation Plan.” (This report explains the applicant’s method for performing the ISV.)

4. APP-OCS-GEH-420, Revision B, “Human Factors Engineering Discrepancy Resolution Process.” (This report explains the applicant’s method for resolving HEDs.)

5. APP-OCS-GEH-520, Revision B, “AP1000 Plant Startup HFE Design Verification Plan.” (This report explains the applicant’s method for verifying the as-built design.)

Based on guidance provided in the NUREG-0800 Section 14.3 and Branch Technical Position HICB-16 (Guidance on the level of detail required for design certification application under 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants") the staff has concluded that all Tier 2* material associated with Chapter 18 will revert to Tier 2 after the plant first achieves full-power operation. This includes Tier 2* information identified in NUREG-1793. This is a change from how HFE-related Tier 2* material was addressed for the previously approved DCD Revision 15 where there was no expiration date. 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," Sections VIII.B.6.b and VIII.B.6.c describe regulatory requirements associated with Tier 2* material and will be amended to reflect this change.

The additional documents were identified as Tier 2* because they describe the specific process the applicant will use to accomplish the final HFE design. The staff has verified that this process conforms to the regulatory guidance in NUREG-0737, which in turn supports the staff's conclusions that the HFE design provides reasonable assurance the Control Room staff can safely control plant operations via the HSIs. These documents also include acceptance criteria that will be used when inspecting the final HFE design conforms to Table 3.2-1 ITAAC.

The applicant responded to RAI-SRP18-COLP-23 R3, dated August 2, 2010, revising Tier 1 and Tier 2 of the DCD to reflect these additional Tier 2* references. In addition, the introduction of the DCD addresses Tier 2* references and specifies when the * designation expires in Table 1-1. The applicant responded to RAI-SRP18-COLP-54, dated August 18, 2010, revising the table to include the additional references listed above and to note all HFE Tier 2* references will expire when the COL holder first achieves 100 percent power operation. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19. SEVERE ACCIDENTS

19.0 Background

In December 2005, the U.S. Nuclear Regulatory Commission (NRC) issued Supplement 1 to NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." NUREG-1793 and its supplement documented the basis for certifying the AP1000 design. Subsequently, the agency has issued new or revised requirements and guidance for addressing severe accidents in the following documents:

Title 10 of the Code of Federal Regulations (10 CFR) Part 52

The Commission issued 10 CFR Part 52, ""Licenses, certifications, and approvals for nuclear power plants," on April 18, 1989. This rule provides for issuing early site permits, standard design certifications (DCs), and combined licenses (COLs) with conditions for nuclear power reactors. It details the review procedures and licensing requirements for applications for these new permits, certifications, and licenses. It is intended to achieve the early resolution of licensing issues, as well as to enhance the safety and reliability of nuclear power plants.

The NRC revised the rule on August 28, 2007. Specifically, 10 CFR 52.47, "Contents of applications; technical information," now requires an application for a DC to describe the design-specific probabilistic risk assessment (PRA) and its results. 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," now requires each COL applicant to describe the plant-specific PRA and its results.

Regulatory Guide 1.200

Regulatory guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,", was issued in February 2004, then revised in January 2007 and March 2009. It describes one acceptable approach for determining whether the quality of PRA provides sufficient confidence in the results to support regulatory decision making for light-water reactors. RG 1.200 endorses, with certain restrictions, a standard published by the American Society of Mechanical Engineers (ASME); ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," including Addenda A and B. It also endorses Nuclear Energy Institute (NEI) 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance."

Interim Staff Guidance

DC/COL-ISG-1, "Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion in Design Certification and Combined License Applications," which clarified the implementation of a performance-based approach for determining site-specific ground motion and methodology for evaluating the effects of high frequency ground motion.

DC/COL-ISG-3, "Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications," clarifies the expectations of the staff with respect to the level of detail to be described and results to be reported in applications.

DC/COL-ISG-20, “Interim Staff Guidance on Implementation of a Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment,” provides more detail on the seismic margin analysis and identifies what to document at the time of application for a COL, and prior to operation.

19.1 Probabilistic Risk Assessment

19.1.1 Introduction

Westinghouse Electric Company, LLC (Westinghouse or the applicant) filed an application for an amendment to the AP1000 DC rule (10 CFR Part 52, Appendix D, “Design Certification Rule for the AP1000 Design”) and provided a revised design control document (DCD). Westinghouse had submitted a design-specific PRA of the AP1000 design as part of the AP1000 design documentation for the certified design. Westinghouse did not submit a revised PRA report with the amendment request; however, Westinghouse did describe, in a number of technical reports (TRs), the changes to the PRA that would result from the design modifications proposed in the amendment. The proposed changes to the DCD reflect these modifications.

The NRC’s regulations at 10 CFR Part 52 no longer require submittal of the PRA report. Instead, applicants are to provide, in the DCD, a description of the PRA and a summary of its results. The design-specific PRA is available for the staff’s review and still forms the basis for the site-specific, plant-specific PRAs that COL applicants must describe and COL licensees must upgrade and update before loading fuel.

Since certification of the AP1000 design, Westinghouse upgraded the PRA as part of a conversion from proprietary software to a widely used program, the Computer-Aided Fault-Tree Analysis System (CAFTA). The PRA was also updated to reflect proposed design changes. As part of the AP1000 DC amendment application, it reported all resulting changes to the insights, assumptions, and results of the analysis.

In addition to the revised DCD, the staff reviewed the following AP1000 COL standard TRs:

- Westinghouse Commercial Atomic Power (WCAP)-16555, APP-GW-GL-011, “AP1000 Identification of Critical Human Actions and Risk Important Tasks,” March 2006

- TR-6, APP-GW-GLR-021, “AP1000 As-built COL Information Items,” June 2006

- TR-34, APP-GW-GLN-016, “AP1000 Licensing Design Change Document for Generic Reactor Coolant Pump,” November 2006

- TR-36, APP-GW-GLR-016, “AP1000 Pressurizer Design,” May 2006

- TR-66, APP-GW-GLR-070, “Development of Severe Accident Management Guidance,” January 2007

- TR-88, APP-GW-GLR-065, “AP1000 Instrumentation & Control (I&C) Data Communication and Manual Control of Safety Systems and Components,” Revision 1, May 2009

- TR-97, APP-GW-GLN-022, Revision 1, “DAS [diverse actuation system] Platform Technology and Remote Indication Change,” May 2007

- TR-101, APP-GW-GLR-101, “AP1000 Probabilistic Risk Assessment Site-Specific Considerations,” Revision 1, October 2007

- TR-102, APP-GW-GLR-102, “AP1000 PRA Update Report,” Revision 1, November 2009

- TR-105, APP-GW-GLN-105, Revision 2, “Building and Structure Configuration, Layout and General Arrangement Design Updates,” October 2007

- TR-106, APP-GW-GLN-106, Revision 1, “Mechanical System and Component Design Update,” September 2007

- TR-130, APP-GW-GLR-130, “Editorial Format Changes Related to Combined License Applicant and Combined License Information Items,” June 2007

- TR-134, APP-GW-GLR-134, Revision 5, “AP1000 DCD Impacts to Support COLA Standardization,” June 2008

- TR-135, APP-PRA-GER-001, “AP1000 Design Change Proposal Review for PRA and Severe Accident Impact,” Revision 1, December 1, 2009

- TR-147, APP-GW-GLN-147, Revision 1, “AP1000 CR and IRWST Screen Design,” March 2008

This information is generic to the design and applies to all COL applications that reference the AP1000 DC.

19.1.1.1 Background and NRC Review Objectives

The general objectives of the NRC’s review of the most recent AP1000 DCD revision include the following:

- identification of new risk-informed safety insights based on systematic evaluations of risk associated with the amended design

- confirmation that regulatory treatment of nonsafety systems (RTNSS) remains appropriate

- confirmation that the DC requirements, such as inspection, tests, analyses, and acceptance criteria (ITAAC), design reliability assurance program (D-RAP), and technical specifications, as well as COL and interface requirements, are amended as appropriate

- confirmation that the conclusions reached in the previous certification remain valid

During the construction stage, the COL applicant will ensure that detailed design documents are consistent with the certified design so that the key assumptions and risk insights from the PRA remain valid. (These assumptions and insights are documented in DCD Table 19.59-18. The D-RAP ITAAC confirm this by ensuring that appropriate quality controls have been applied in the development of detailed design for procurement and construction.) The COL applicant will ensure, through other ITAAC and preoperational programs, that the configuration of the plant, as built, is consistent with the detailed design. The Commission believes that updated PRA insights, if properly evaluated and used, could strengthen programs and activities in areas such as training, development of emergency operating procedures (EOPs), reliability assurance, maintenance, and evaluations performed pursuant to 10 CFR 50.59, "Changes, tests and experiments." The design-specific PRA, developed as part of the DC process, should be revised to account for site-specific information, as-built (plant-specific) information refinements in the level of design detail, technical specifications, plant-specific EOPs, and design changes. The COL licensee is responsible for these updates. This is part of COL Information Item 19.59.10-2.

The NRC requires the COL applicant to develop a plant-specific PRA based on the design-specific PRA. At the time of application, the plant-specific PRA of internal events (both at power and shutdown) must address, at a minimum, proposed deviations from the certified design. The plant-specific PRA of external events must evaluate external events applicable to the proposed site and confirm that they are bounded by the PRA. This is also part of COL Information Item 19.59.10-2. The staff expects that the COL applicant and licensee will use the plant-specific PRA and revised failure rates (when available) to update, as appropriate, its reliability assurance programs (including the quality assurance program and the maintenance rule program).

19.1.1.2 Evaluation of Probabilistic Risk Assessment Quality and Closure of Open Issues

The NRC staff evaluated the information submitted by the applicant in accordance with NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." For Chapter 19, the staff used Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," and Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

In TR-102, the applicant described conversion of the PRA modeling software package from its proprietary program, WesSAGE, to the more widely used program, CAFTA. The applicant also updated the AP1000 PRA to include the most recent instrumentation and controls (I&C) design information. The applicant documented the basis for its determination that structures, systems, and components (SSCs) modeled in the PRA were not affected by other design changes in a manner that affected the PRA.

In its original review of the AP1000 PRA, the staff relied on the similarity between the AP600 and AP1000 certified designs to reduce the review effort. This similarity (e.g., in system design and overall plant layout) allowed the use of the AP600 PRA as the starting point in the development of the AP1000 PRA. Similarly, the PRA associated with the currently certified AP1000 design was the starting point for upgrading and updating the AP1000 PRA in support of the DC amendment. In addition to reviewing the description of changes to the PRA, the staff reviewed the description of the new I&C design, specifically the plant control system (PLS) and the protection and safety monitoring system (PMS).

The staff used reported PRA results, as well as the results of sensitivity, uncertainty, and importance analyses, to focus its review. The staff also used applicable insights from previous PRA studies regarding key parameters and design features.

The review of the quality and completeness of the AP1000 PRA included the issuance of several requests for additional information (RAIs) to the applicant related to TR-102. Following its review of the responses to the RAIs, in August 2007, the staff conducted an audit at the applicant's offices, focusing on three principal areas:

1. assessment of the applicant's process to upgrade the PRA model, including the process by which they assessed the design and operational changes for potential impact on the PRA

2. review of the changes to the model since the applicant submitted the previous PRA report, especially those resulting from proposed changes to the certified design

3. inspection of the model itself to confirm that the model accurately reflected modifications to the design and that the model is a suitable basis for PRAs required of COL applicants that reference the AP1000 DC

In the process, the staff reviewed the qualifications of personnel involved in PRA-related activities and found them to be acceptable. The staff also examined the procedures the applicant used to review modifications for their potential impact on the PRA, to identify and correct problems in the model, to implement changes to the model, and to assess the results of analysis. During this review, the staff developed further requests for additional information based on NUREG-0800 Chapter 19, discussed below.

Section 19.1.10 of this report provides a summary and the resolution of open items. These are issues resulting from the review of Chapter 19 that had not been resolved when the draft of this report was prepared for review by the Advisory Committee on Reactor Safeguards.

19.1.2 Special Advanced Design Features

19.1.2.1 Special Advanced Design Features for Preventing Core Damage

The applicant proposed changes to the certified AP1000 design that have the potential to affect the PRA. The following sections discuss these changes.

19.1.2.1.2 Defense-In-Depth Active Nonsafety-Related Systems

The AP1000 design incorporates several active systems that are capable of performing some of the same functions as those performed by the safety-related passive systems. The availability of such redundant systems minimizes the challenge to the safety-related passive systems by providing core cooling during normal plant shutdowns and serving as a first line of defense during accidents.

The diverse actuation system (DAS) provides an alternate means for initiating automatic and manual reactor trip and actuation of selected engineered safety features that is diverse from the safety-related PMS. An additional DAS squib valve control cabinet, spatially separated from the DAS cabinet in the control room, provides additional confidence that operators can take manual actions to depressurize the reactor coolant system (RCS) and initiate key functions, such as in-containment refueling water storage tank (IRWST) injection, containment recirculation, and IRWST drain to containment. In DCD Table 19.59-18, the applicant clarified the degree of diversity between the PMS and the DAS. Section 7.7 of this report includes the staff evaluation of this modification. In RAI-SRP19.0-SPLA-06, the staff requested information on the potential of this modification to affect the timing of steps taken to mitigate an anticipated transient without scram (ATWS) event (positively or negatively) and to reduce risk by providing a spatially diverse actuation station.

In a letter dated August 21, 2008, the applicant responded that the addition of the remote DAS cabinet would not result in a significant change to the risk importance of any SSC or human action. The probability of using this remote cabinet is very low, as it would require several very unlikely events to occur. An example of such a situation would be the need to use manual controls coupled with the need to evacuate the control room and the failure or unavailability of the remote PMS panels. It is very likely that a fire or event that would make the control room and the remote PMS panel unavailable for manual operation would also result in a successful automatic shutdown of the AP1000. In addition, human reliability analyses (HRAs) for the DAS use values that represent low probability of success, high-stress situations. This supports the conclusion that the modification would not result in a significant change to the risk importance of any SSC or human action. For that reason, the AP1000 PRA does not model the use of this remote panel separately from use of the one in the control room.

The staff noted that the PRA models manual action as a single basic event irrespective of the location from which that action is taken. Furthermore, the proposed modification reduces uncertainty in the performance of the action. Because the need to use the remote DAS cabinet requires multiple, simultaneous, and highly unlikely events, the risk importance of SSCs or human actions would not be significantly altered by additional detail in the model. For these reasons, the staff concludes that the applicant's decision not to alter the modeling of this system is conservative and acceptable. The staff considers RAI-SRP19.0-SPLA-06 resolved.

19.1.2.1.7 Redundant Long-Term Recirculation Systems

RCS recirculation is required for long-term core cooling during loss-of-coolant accidents (LOCAs) and whenever the feed-and-bleed method is used to cool the core during an accident. In the AP1000, recirculation can be achieved either by gravity (through the safety-related IRWST injection lines) or pumping (through the nonsafety-related normal residual heat removal system (RNS)) with suction from the containment sump. Two redundant recirculation lines exist (one for each of the two redundant IRWST injection lines). Furthermore, each recirculation line has two paths that are redundant, with the exception of the recirculation screens. Though there are two separate screens, the applicant does not characterize them as redundant and explicitly models their common-cause failures (CCFs). Section 6.2.1.8 of this report documents the staff's evaluation of the screen design.

In RAI-SRP19.0-SPLA-04, the staff asked the applicant to discuss the impact of changes to the design of the recirculation system on the results and insights of the shutdown risk assessment.

In a letter dated July 22, 2008, the applicant stated that the structural integrity of the new recirculation screen configuration exceeds that of the screen-like material typically used in current pressurized-water reactor (PWR) sump screens, precluding the need for trash racks. Screen testing demonstrated that the new screens will not experience a significant head loss while operating within design-basis flow/debris conditions. A qualitative assessment concluded that the enhancement will reduce failure probability, while the increased flow area will improve accident response.

The staff finds it reasonable to expect that the proposed modification will improve performance of the recirculation screens as compared to the certified design. Although the large, interconnected screens are not independent, the staff finds that the applicant adequately addressed CCF of the screens. For these reasons, the staff concludes that the applicant's decision not to alter the modeling of this system is conservative and acceptable. The staff considers RAI-SRP19.0-SPLA-04 resolved.

19.1.2.1.9 Canned Reactor Coolant Pumps

The AP1000 design originally specified canned reactor coolant pumps (RCPs). In TR-34, the applicant specified sealless RCPs, which may be canned-motor or wet-winding pumps. For both canned-motor and wet-winding pumps, the motor and all rotating components are inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, all of which are designed for full RCS pressure. Because the rotor and shaft connecting it to the impeller are contained within the pressure boundary, a seal is not required to restrict leakage out of the pump into containment. In addition, the heat exchanger that cools the RCP has been modified; it is now external to the pump. The applicant asserts that these changes do not alter the PRA model.

Section 5.4 of this report discusses the staff's evaluation of changes to the RCP design. The pump design is important because the use of sealless RCPs in the AP1000 design eliminates the RCP seal LOCA (an important contributor to risk for most operating commercial PWRs). In addition, water is used to lubricate and remove heat from pump bearings, eliminating the need for RCP lubricating oil systems and the attendant fire hazard. Because the proposed design alternative of a wet-winding rotor changes neither the failure modes of the RCP and its heat exchanger nor the estimated reliability of these components, the staff concludes that no change to the internal events PRA model is required. However, this change had an impact on the seismic margin analysis. The staff's evaluation of this issue is addressed in Section 19.1.5.1, "Probabilistic-Risk-Assessment–Based Seismic Margin Analysis."

19.1.2.1.10 Improved Control Room Design and Digital Instrumentation and Control Systems

The AP1000 control room is an advanced design that is expected to provide information that is presented to the operator in a way that is more easily used than the displays in currently operating reactor designs. Similarly, control is expected to be easy and consistent, and nearly all actions can be performed from a single station. Section 7.1.4 of this report documents the staff's review of the AP1000 control room design.

The PRA took no credit for the impact of the advanced control room on normal operations and emergency response (e.g., initiating event frequency or HRA). Because the impact of an advanced control room is still the subject of research and control room design verification and validation cannot be performed until a control room is simulated, the staff concludes that this approach is conservative and acceptable for the DC.

During the August 2007 audit of the applicant's PRA, the staff identified a discrepancy in the CCF probability of PMS component interface modules for the recirculation squib valve (V-118). The applicant immediately initiated corrective action. In RAI-SRP19.0-SPLA-07, the staff requested correction of the discrepancy and an updated report of PRA results. Specifics requested included: (1) the results of re-solving and re-quantifying the baseline and RTNSS full-power PRA, the shutdown PRA, and the external events PRA; (2) new risk insights identified during requantification of the previously mentioned PRAs; and (3) the results of the revised importance analysis.

In a letter dated August 21, 2008, the applicant reported that, in addition to correcting the discrepancy, analysts identified measures to improve the realism of the PRA for I&C systems. Specifically, the analyses found the values selected for PMS and PLS component common-cause beta factors to be overly conservative. The applicant has revised the model to reflect component-specific common-cause beta factors for PMS and PLS system components modeled in the PRA.

In the same letter, the applicant committed to revising TR-102 to reflect: (1) the results of re-solving and re-quantifying the baseline and RTNSS full-power PRA, the shutdown PRA, and the external events PRA; (2) any new risk insights identified during requantification of the previously mentioned PRAs; and (3) the results of the revised importance analyses. The applicant also reported that “requantification of the at-power PRA indicate[s] that the core damage frequency (CDF) and large release frequency (LRF) values and top cutsets closely compare with these items documented in the [previously submitted Level 1 internal events] PRA….”

In a letter dated November 6, 2008, the applicant reported some results of model correction and requantification. (The software used for this process automatically re-solves the model each time the model is requantified.) The letter reported only those results of the PRA that were point estimates of CDF and LRF (total, at power, shutdown, and sensitivity to nonsafety SSCs). The applicant reported no other changes in risk insights or importance analysis. The staff noted several changes that met the criteria of DC/COL-ISG-3 during the October 2009 onsite audit of the PRA. For example, the applicant had not reported a shutdown PRA sequence of significance. In another case, design improvements had eliminated a risk-significant component. Because these and similar changes were not reflected, the staff did not find this letter to be fully responsive. The staff identified the absence of some corrected results in the DCD as Open Item OI-SRP19.0-SPLA-07.

In a letter dated December 17, 2009, the applicant proposed revisions to the DCD. The staff’s evaluation of these proposed revisions and resolution of the open item is documented in Section 19.1.4.1, “Level 1 Shutdown Internal Events Probabilistic Risk Assessment.”

19.1.2.1.11 Large Pressurizer and Low-Power Density

The AP1000 pressurizer is large in comparison to the pressurizer of currently operating plants. This reduces the frequency of reactor scrams by increasing transient operation margins. This feature also moderates the pressure rise during certain transient events, such as loss of main feedwater, thus reducing the likelihood of a challenge to the primary safety valves. A larger pressurizer volume, as compared to currently operating plants, also helps lower the peak pressure that can be reached after a postulated ATWS event.

The applicant found it necessary to alter the design of the pressurizer. TR-36 details this change. Section 5.4.5 of this report documents the staff’s evaluation of this design change. The applicant did not propose a change to the PRA because of this modification.

Because the applicant analyzed the proposed design changes to the pressurizer and found that they do not alter system-level thermal-hydraulic response or success criteria, the staff concludes that no change to the internal events PRA model is necessary. However, this change had an impact on the seismic margin analysis. The staff's evaluation of this issue is addressed in Section 19.1.5.1, "Probabilistic-Risk-Assessment–Based Seismic Margin Analysis."

19.1.2.2 Special Advanced Design Features for Core Damage Consequence Mitigation

The following design features improve the ability of the containment to accommodate the challenges associated with severe core damage accidents. The AP1000 PRA and supporting deterministic analyses model the impact of these features on severe accident mitigation and containment performance.

19.1.2.2.4 External Reactor Vessel Cooling

To accommodate the higher decay heat level in the AP1000, Westinghouse needed to refine the AP600 reactor vessel insulation system (RVIS). The design was modified to increase the critical heat flux (CHF) at the surface of the reactor pressure vessel (RPV), enhancing the heat transfer through the RPV to the surrounding water. TR-24, APP-GW-GLR-060, "Reactor Vessel Insulation System - Verification of In-Vessel Retention Design Bases," February 2007, addresses COL Information Item 5.3-5 by verifying that reactor vessel insulation is consistent with the design bases established for in-vessel retention of a damaged core. COL Information Item 5.3-4 requires a structural analysis of the AP1000 reactor vessel insulation and support structure. TR-24 reports relevant results of that analysis.

The effectiveness of external reactor vessel cooling in the AP1000 design depends, in part, on a RVIS that provides an engineered pathway for supplying water cooling to the vessel exterior and venting steam from the reactor cavity during severe accidents. It is designed to limit thermal losses during normal operations. Section 5.3 of this report documents this design, which is discussed in Section 19.1.8.24 and evaluated in Section 19.2.3.3.1.3.2.

In RAI-TR24-SPLA-06, the staff noted that some paints and coatings used to protect the reactor vessel during shipping could have detrimental effects on CHF performance. In TR-106, the applicant stated that the external surface of the reactor vessel is bare metal. AP1000 DCD Section 5.3.4.5 now reflects the fact that a temporary protective coating applied before shipment will protect carbon steel surfaces. In DCD Section 19.34.2.1, the applicant stated that the vessel will have no coatings on the outside surface of the reactor vessel. This ensures that wetability of the surface will not be inhibited and CHF performance will not be degraded; the staff finds this acceptable. The COL licensee must remove these temporary coatings. The staff requested additional basis for confidence that this will be accomplished.

In a letter dated April 14, 2009, the applicant clarified the nature of the protective covering, which is to be an industrial form of shrink wrap that will be removed in the receiving process. The staff agrees that no additional controls are required; the statement in DCD Section 19.34.2.1 is sufficient and RAI-TR24-SPLA-06 is resolved.

19.1.2.3 Residual Risk from Changes Not Explicitly Modeled

The applicant reviewed all design changes for their potential to affect risk. TR-135 documented the process used for this review as well as the results of that process. The staff noted that, if a design change proposal (DCP) dealt with an SSC modeled in the PRA, the applicant evaluated its potential to affect the PRA results. However, the applicant did not necessarily evaluate other changes that may have an impact (e.g., changes to assumptions or PRA insights, as well as changes to model logic or changes that may alter probabilistic parameter estimates).

For example, a new or revised operating procedure might alter, for some modes, the alignment of an SSC in a manner that is inconsistent with documented insights or assumptions. The equipment would then require realignment to prevent or mitigate the consequences of an event applicable to the mode in question. It may be appropriate to model a different basic event (and supporting SSCs) in the PRA model for that mode. Alternatively, additional constraints or conditions to control risk may be appropriate before initiating the proposed procedure. The staff expects the applicant to perform such assessments, even if it will usually result in a determination that no explicit model change or procedural constraint is necessary.

In RAI-SRP19.0-SPLA-05, the staff identified the specific example of vacuum fill operation and asked the following:

> Please identify and briefly describe each DCP incorporated in the amended design and assess its potential to have such an impact on the PRA. For each DCP that may have an impact upon the PRA or other risk studies (e.g., seismic and internal fire) please evaluate and report its potential significance.

In a letter dated September 5, 2008, the applicant provided the results of its evaluation of the vacuum fill operation. In addition, the applicant reported that it made a change in the process used for future DCPs. The applicant reviewed each one for its impact on the PRA, with no initial screening for PRA-modeled SSCs. This effort included a documented review of the PRA assumptions and PRA insights affected, potential changes to model logic and probability data, the effect of operational changes on component modeling, and the impact on other risk studies (e.g., seismic and internal fire). The documentation of the review identifies and briefly describes every DCP and provides the rationale used to determine its impact on the plant risk and changes to the PRA or other risk studies (e.g., seismic and internal fire).

The staff concludes that the applicant performed an appropriate evaluation of the risk implications of vacuum fill operations and that the design change process will provide adequate assurance that the risk implications of all changes after Revision 17 of the DCD will be assessed. In a letter dated April 23, 2009, the applicant provided a schedule for re-evaluation (using the revised criteria) of all DCPs processed to date: changes reflected in Revision 17 of the DCD will be re-evaluated first and re-evaluation of all earlier changes will be documented prior to initial fuel loading. The results will be reflected in the plant-specific PRA as upgraded and updated prior to initial fuel load. The COL licensee is responsible for these updates as part of COL Information Item 19.59.10-2. The staff finds that this provides adequate assurance that the risk implications of all changes will be appropriately assessed and, if necessary, analyzed. This is an acceptable method for controlling residual risk from changes that are not explicitly modeled in the internal events PRA. The staff considers RAI-SRP19.0-SPLA-05 resolved.

19.1.3 Safety Insights from the Internal Events Risk Analysis (Operation at Power)

Safety insights from the internal events Level 1 PRA include the following:

- dominant accident sequences contributing to CDF

- areas in which certain AP1000 design passive and defense-in-depth features were the most effective in reducing risk as compared to currently operating reactor designs
- major contributors to the estimated CDF from internal events, such as hardware failures, system unavailabilities, and human errors
- major contributors to maintaining the built-in plant safety (to ensure that risk does not increase unacceptably)
- major contributors to the uncertainty associated with the estimated CDF
- sensitivity of the estimated CDF from internal events to: (1) potential biases in numerical values; (2) assumptions made; (3) lack of modeling details in certain areas; and (4) previously raised safety issues

Safety insights from the internal events Level 2 PRA include the following:

- core damage sequences and accident classes contributing to containment failure
- frequency and conditional probability of containment failure
- leading contributors to containment failure and risk

19.1.3.1 Level 1 Internal Events Probabilistic Risk Assessment

In TR-102, the applicant described the results of the PRA that it had upgraded and updated to conform to the amended design.

The staff conducted an audit of the PRA model upgrade and update. The staff reviewed the qualification of the PRA staff, procedures used for conversion, and processes for updating the PRA. In addition, the staff examined the PRA model itself with emphasis on new fault trees developed for I&C systems. Chapter 7 of this report documents the staff's review of I&C system design changes. The staff also reviewed the electrical system model changes for consistency with Revision 1 of TR-79. Chapter 8 of this report documents the staff's review of electrical system design changes.

The staff found that the development of the I&C model was consistent with the amended I&C design, as described in TR-39, APP-GW-GLN-004, "Instrumentation and Control Design Change," May 2006; WCAP-16675-NP, APP-GW-GLR-071, "AP1000 Protection and Safety Monitoring System Architecture Technical Report," Revision 2, May 2009; APP-GW-GLR-018, "Failure Modes and Effects Analysis and Software Hazards Analysis for AP1000 Protection System," June 2006; and TR-97, Revision 1. TR-80, APP-GW-GLR-080, "Mark-up of AP1000 Design Control Document Chapter 7," October 2007, documents the impact of these design changes on the DCD. Chapter 7 of this report documents the staff's review of the I&C design changes. (Many of the design changes had no impact on the PRA and, therefore, no impact on the severe accident analysis, as documented in TR-102 and TR-135.)

In a letter dated November 6, 2008, the applicant reported some results of the PRA model requantification. The applicant estimated the mean CDF for the AP1000 design from internal events during operation at power to be about 2.41×10^{-7} per year, unchanged from what the previous PRA reported. The applicant characterized this as equivalent, given appropriate treatment of uncertainties.

Although the applicant did not modify the initiating event frequencies in the model, the contribution of each initiating event to CDF changed slightly. The applicant reported that these changes were associated with the I&C model revision and updated electrical power dependencies. The applicant's assessment suggested that the changes were not of sufficient magnitude to alter the risk insights derived from the PRA results. For example, the top ten cutsets were identical, and the CDF attributable to failure of the most risk-significant system (the PMS) changed by only a small factor (i.e., it became about half as significant). Various LOCA initiating events continue to dominate the CDF profile (about 85 percent), followed by reactor vessel rupture (about 4 percent) and transient events (about 4 percent). Contributions from steam generator tube rupture (SGTR) events are slightly higher (about 4 percent), while ATWS sequences and loss of offsite power/station blackout events contribute even less than before (less than 1 percent).

Based on these results and the audit that provided confidence in the model upgrade and update process, the staff finds that the amended Level 1 internal events PRA at power did not change significantly. The staff finds that a plant-specific PRA report that is identical to the PRA for the certified design continues to provide an acceptable basis for risk insights and assumptions related to internal events.

However, changes to the design have altered some of the insights derived from the PRA (e.g., improving the design by eliminating a risk-significant SSC). As discussed in Section 19.1.2.1.10, in RAI-SRP19.0-SPLA-07, the staff requested the results of re-solving and requantifying the baseline and RTNSS full-power PRA, shutdown PRA, and external events PRA, as well as the results of the revised importance analyses. In a letter dated August 21, 2008, the applicant committed to re-solve and requantify the model after making some corrections. In a revised response dated November 6, 2008, the applicant altered this commitment, as discussed in Section 19.1.2.1.10, above, where it is identified as Open Item OI-SRP19.0-SPLA-07. Resolution of the open item is documented in Section 19.1.4.1 of this report.

19.1.3.2.3.3 Human Actions

In WCAP-16555, the applicant reviewed human actions with respect to risk achievement worth (RAW) and risk reduction worth (RRW). In addition, the applicant reviewed human actions required for maintenance, test, inspection, and surveillance (MTIS) support. The applicant stated that, on a deterministic basis, no human actions were required to mitigate any design-basis accident (DBA) or to prevent core damage following a DBA.

The applicant also identified 19 human actions as most significant from a probabilistic standpoint, though none of them came within an order of magnitude of the criteria previously accepted for a "critical" human action. The applicant added the following three human actions because an expert panel considered them to be significant:

1. Failure to recognize the need for and failure to isolate the RNS system, given rupture of the RNS piping when the plant is at hot/cold conditions (RHN-MAN04): The applicant added this action because of the short time available for the operator to act and the conflicting goals (maintaining core cooling by the RNS versus isolating a leak or break in the RNS piping).

2. Failure to recognize the need for and failure to actuate the hydrogen control system, given core damage following a LOCA (VLN-MAN01): The applicant added this action because its limiting RAW is relatively close to the criteria, and it is a function within the scope of RTNSS.

3. Failure to close equipment hatch and personnel airlocks following core damage during a shutdown event: The applicant added this action because human action importance could not be calculated for shutdown, internal events, or LRF. The expert panel considered that, under these conditions, the largest risk of large release would come from failure to close the containment. Closing the containment under these conditions involves closing the equipment hatch, personnel hatches, and temporary penetrations.

As noted by the staff, DCD Table 19.59-18 documents that it is important to maintain the ability to close containment hatches and penetrations during MODE 5 and MODE 6 before steam is released into the containment. There is a commitment for procedures and training to ensure that this action will be taken when required.

The staff found that the results were consistent with the methodology prescribed for the certified design and that the applicant conservatively identified risk-important human actions. For these reasons, the staff finds the results to be consistent with NUREG-0800 and, therefore, acceptable.

19.1.3.3.3 Important Insights from Level 3 PRA and Supporting Sensitivity Analyses

The applicant deleted the discussion of Level 3 PRA from Tier 2. The Level 3 PRA is now described only in the environmental assessment.

19.1.4 Safety Insights from the Internal Events Risk Analysis for Shutdown Operation

19.1.4.1 Level 1 Shutdown Internal Events Probabilistic Risk Assessment

The staff compared the results of the shutdown PRA, as seen in the current model, with the results reported in Revision 16 of the DCD (unchanged in Revision 17). Many of the results significantly differ from those reported in DCD Section 19.59.5.1, “Summary of Shutdown Level 1 Results.” For example, Section 19.59.5.1 discusses the dominant sequences and key contributors to risk. The staff compared this documentation to the top 15 cutsets and the top 20 component basic events ranked by RAW from the CAFTA results. Loss of component cooling (supplied by the service water system [SWS]) or service water (supplied by the circulating water system [CWS]) during drained conditions contributes at least 73 percent to the CDF, as seen in the CAFTA results, versus 64 percent as reported in the DCD. Loss of the RNS initiating event during drained conditions contributes at least 10 percent to the CDF as compared to 6 percent reported in the DCD. Inadvertent draining through valve V024 (IEV-LOCA24ND) contributes more to the CDF than the risk of RCS overdraining, as seen in the CAFTA results. However, the DCD does not report this event. Some of these changes appear to meet the importance criteria of DC/COL-ISG-3 and, therefore, should be documented.

In RAI-SRP19.0-SPLA-13, the staff asked the applicant to update DCD Table 19.59-15, "Summary of AP1000 Results," and to provide the following information:

1. a list of cutsets for the AP1000 shutdown PRA that contribute to 95 percent of total shutdown CDF and any that contribute as much as 1 percent of total shutdown CDF

2. a list of all SSCs in the shutdown PRA with their RAWs (if RAW greater than 2)

3. a list of all human actions modeled in the shutdown PRA with their RAW

4. a list of all CCFs in the shutdown PRA with their RAW (if RAW greater than 2) (or confirmation that all are described in WCAP-16555)

In a letter dated August 21, 2008, the applicant stated that the next revision of TR-102 would include the PRA model changes discussed in response to RAI-SRP19.0-SPLA-13. In a subsequent letter dated November 6, 2008, the applicant stated that it would revise TR-102 but proposed no changes to the DCD. The NRC staff identified this as the first part of Open Item OI-SRP19.0-SPLA-13.

The staff conducted an audit of the corrected and amended PRA model at the applicant's offices on October 13–15, 2009, as documented in an audit report dated November 22, 2009. The staff determined that identified deficiencies in the PRA model had been corrected. In a letter dated December 17, 2009, the applicant proposed revisions to the DCD to amend the description of shutdown PRA results. The staff finds that the proposed changes to the DCD are consistent with DC/COL-ISG-3 and, therefore, acceptable. Therefore, Open Item OI-SRP19.0-SPLA-07 (addressed in Section 19.1.2.1.10) and Open Item OI-SRP19.0-SPLA-13 are considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19.1.4.2 Dominant Accident Sequences Leading to Core Damage

In RAI-SRP19.0-SPLA-13, the staff also asked the applicant to confirm that the list of major contributors to risk for each sequence that contributes more than 1 percent to the shutdown CDF remains consistent with the cutset results and to revise the DCD as necessary to describe all such sequences.

In the August 21, 2008 letter, the applicant stated that the next revision of TR-102 would include the PRA model changes discussed in this RAI-SRP19.0-SPLA-13 response. In a subsequent letter dated November 6, 2008, the applicant stated that it would revise TR-102 but proposed no changes to the DCD. The staff identified this as the second part of Open Item OI-SRP19.0-SPLA-13.

In a letter dated December 17, 2009, the applicant proposed revisions to the DCD. The staff's evaluation of these proposed revisions and resolution of Open Item OI-SRP19.0-SPLA-13 is documented in Section 19.1.4.1, above.

19.1.4.3 Risk-Important Design Features

The applicant now describes actuation of IRWST injection as the result of a fourth-stage automatic depressurization system (ADS) signal rather than a low hot-leg level signal. In the first part of RAI-SRP19.0-SPLA-04, the staff asked the applicant to clarify the impact of this modification on the shutdown risk assessment.

The applicant responded in a letter dated July 22, 2008, that it had modified the logic description in the DCD to represent more clearly how the system is intended to function during shutdown conditions.

Appendix 19E to DCD Revision 15 described the logic in the AP1000 as follows:

- actuation of IRWST injection on low (empty) hot-leg level on a two-out-of-two basis (RCS hot-leg level channel basis)

- actuation of fourth-stage ADS valves on low (empty) hot-leg level on a two-out-of-two basis (RCS hot-leg level channel basis)

The DCD provides the following revised description:

- actuation of fourth-stage ADS valves on low (empty) hot-leg level on a two-out-of-two basis (RCS hot-leg level channel basis)

- actuation of fourth-stage ADS causes actuation of IRWST injection

This logic configuration forms the basis for the PRA model and shutdown risk assessment. The change in wording for the logic for actuation of IRWST injection has no impact on the results and insights of the shutdown risk assessment.

The staff agrees that the clarification did not alter the functional response of the system and confirmed that the change in description did not affect shutdown PRA insights. The staff considers the first part of RAI-SRP19.0-SPLA-04 resolved.

COL Information Item 18.7-1 includes the following statement:

> Since inadvertent opening of RNS valve V024 results in a draindown of RCS inventory to the IRWST and requires gravity injection from the IRWST, the COL applicant will have administrative controls to ensure that inadvertent opening of this valve is unlikely. The control room design will take into account this error.

In RAI-SRP19.0-SPLA-09, the staff requested information on the features of the control room design that will ensure that inadvertent opening of valve V024 is unlikely.

In a letter dated August 21, 2008, the applicant responded that, during shutdown, electrical power to valve V024 is blocked (breakers open and manually locked out) when RNS is in operation. This prevents inadvertent operation when the RCS would be depressurized. (ADS valves are open during shutdown conditions in accordance with Technical Specification 3.4.13.)

A permissive signal (valves V001A/B and V002A/B are fully closed and valve V023 is open) is required to permit manual opening of this valve V024. DCD Figure 7.2-1 shows a corresponding interlock to open the RNS hot-leg suction isolation valves, which is prevented if the IRWST cross-connects to the RNS (valves V023 and V024) are not fully closed.

The staff finds this to be an acceptable method of ensuring that inadvertent opening of a valve V024 is unlikely. The staff considers this portion of COL Information Item 18.7-1 to be closed and RAI-SRP19.0-SPLA-09 resolved.

19.1.4.3.2 Loss-of-Coolant Accidents during Safe Shutdown or Cold Shutdown or Both with the Reactor Coolant System Intact

The applicant modified the containment recirculation design to provide large, interconnected screens without separate trash racks or coarse and fine screens. Section 6.2.1.8 of this report documents the staff's assessment of this change. In the second part of RAI-SRP19.0-SPLA-04, the staff asked the applicant to clarify the impact of this modification on the shutdown risk assessment.

The applicant, in a response dated July 22, 2008, stated that the DCD reflects the use of large, interconnected recirculation screens for recirculation flow. The passive core cooling system (PXS) has two banks of interconnected screens that filter recirculation flow. The staff evaluated these screens using the guidance in RG 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003. The screens are constructed of perforated stainless steel plate that is used to form pockets. Actions taken to close Generic Safety Issue 191, "Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation," May 2005, and to respond to generic letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004, resulted in more robust sump screen designs. As is the case with current operating plants, the structural integrity of the AP1000 screens precludes the need for trash racks. TR-147 also discusses the design of the AP1000 screens. Section 6.2.1.8 of this report documents the staff's evaluation of the screens.

The applicant judged the changes in the screen design to have no negative impact upon the PRA results; thus, the DCD PRA was not changed to reflect modifications to the screens. The applicant judged these changes to have a positive impact upon the PRA, with a lower failure probability of the screens resulting from the enhanced design and increased flow area. The DCD PRA did not credit this change in failure probability.

The staff confirmed that the PRA appropriately modeled the CCF of the screens, using a conservatively large beta factor. For the DC amendment, the modeling is conservative and, therefore, acceptable to the staff. The staff considers the second part of RAI-SRP19.0-SPLA-04 resolved.

19.1.4.3.6 Loss of the Normal Residual Heat Removal System (due to Loss-of-Coolant Accidents) or Loss of the Normal Residual Heat Removal System or Its Support Systems during Reactor Coolant System Open Conditions

When the RCS is open, the importance of the RNS is higher than at other times because safety-related heat removal paths may not be available. An external event (high winds) can have an impact on alternating current power sources required for RNS function because those sources (and their fuel) are not protected by safety-related structures. In addition, if the RNS becomes unavailable, the containment must be closed before boiling begins in the RCS.

In RAI-SRP19.0-SPLA-18, the staff asked the applicant to evaluate high winds while in MODE 5 and MODE 6 or to provide an acceptable basis for screening such events from consideration.

The associated risks should be quantified and possibly controlled. The NRC staff identified this as OI-SRP19.0-SPLA-18.

In a letter dated March 26, 2009, the applicant addressed high wind events occurring in MODE 5 and MODE 6. The applicant stated that emergency response requirements or emergency action levels will require that the RCS be taken out of mid-loop operation and prohibit entry when a potentially severe high wind event is anticipated. In addition, the response describes how core cooling is accomplished if diesel generators are not available.

The staff finds that the proposed measures are appropriate and sufficient to justify screening high wind events during MODE 5 and MODE 6 from further analysis. Controls on the implementation of emergency response requirements are an acceptable way to ensure that these measures will be implemented. The staff considers Open Item OI-SRP19.0-SPLA-18 to be closed.

19.1.5 Safety Insights from the External Events Risk Analysis

Three sections of the AP1000 DCD address PRA of external events consistent with DCD Section 1.9.5.2.14:

1. A risk-based seismic margin analysis (SMA), documented in DCD Section 19.55 and Appendix 19A, both titled “Seismic Margin Analysis,” addresses seismic events. Sections 19.1.5.1 of this report document the staff’s evaluation of the SMA.

2. APP-GW-GL-022, “AP1000 Probabilistic Risk Assessment,” of July 2004, Revision 8, Chapter 57, “Fire Risk Assessment,” documents analysis of the risk associated with internal fires. (The analysis is not discussed in this report because it has not changed since initial certification of the AP1000 design.)

3. DCD Section 19.58, “Winds, Floods, and Other External Events,” addresses remaining external events. Sections 19.1.5.4 through 19.1.5.7 of this report document the staff’s evaluation.

The objectives of the external events risk analysis provided in Section 19.58 of the AP1000 DCD are threefold:

1. Determine screening criteria and identify potential external events that may affect the AP1000 risk on a site-specific basis.

2. Provide generic risk analyses, based on bounding assumptions regarding site-specific parameters (e.g., frequency of each category of hurricanes) for relevant external events.

3. Provide guidance to COL applicants regarding the verification of the applicability of these generic analyses to a specific site.

The AP1000 DCD addresses those external initiating events or external hazards whose causes are external to the plant, other than seismic events. Based on the modified individual plant examinations of external events (IPEEE) guidelines, DCD Section 19.58 discusses the following external events or external hazards:

- high winds (including tornadoes)

- external floods
- external fires
- transportation and nearby facility accidents

The scope of this analysis does not include sabotage, which is consistent with the SRP and therefore acceptable to the staff. The information provided in DCD Section 19.58 is based primarily on the following:

- NRC guidance for the preparation and submittal of IPEEE for operating nuclear power plants

- the AP1000 DC PRA

- site-specific information related to external events for several proposed sites to build a nuclear plant referencing the AP1000 design

On June 28, 1991, the NRC issued Supplement 4 to GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," requesting that each licensee conduct an IPEEE. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events for Severe Accident Vulnerabilities," documents NRC guidelines for conducting IPEEE and on the structure and content of the IPEEE submittal. The staff examined these guidelines to verify their applicability to new reactor licensing and to investigate their completeness. The staff concludes that the IPEEE guidelines are applicable to the process of COL application after they are properly modified as follows:

- The IPEEE, performed by operating reactor licensees, takes into account plant-specific licensing information regarding external hazards that is not necessarily available to a COL applicant. For example, NUREG-1407 states, "[the] effects of external fires, other than loss of offsite power (LOSP), have been evaluated during the operating license (OL) review against sufficiently conservative criteria." Thus, the assumptions made in NUREG-1407 (e.g., in deriving the list of external events to be included in the IPEEE submittal) must be examined to determine whether a COL applicant must address events that were not included in the DCD.

- The baseline risks of the AP1000 design, as assessed in the DC PRA, are lower than the corresponding risks of an average operating plant. At operating reactors, the combined CDF for external events may be of the same magnitude as CDF for internal events. For this reason, the criteria for screening external events from the quantitative evaluation must be properly adjusted to maintain the conclusion reached in the DC—that the AP1000 design represents a reduction in risk compared to existing plants.

The applicant gathered site-specific external events information from utilities interested in the AP1000 design and performed a generic analysis for each external event, based on the most limiting parameters from any site. In TR-101, the applicant identified potential external events that may affect the AP1000 risk.

The staff finds that these external hazards are most likely a complete list of external events associated with candidate sites for an AP1000 plant as of the date of this report. The staff evaluated the analysis of the AP1000 response to external events according to the SRP, which states that the applicant's analyses should be "comprehensive in scope and address all applicable...external events and all plant operating modes." The staff requested additional information in RAI-TR101-SPLA-01 through RAI-TR101-SPLA-08 to clarify the report. In a letter dated October 19, 2007, the applicant responded. The staff requested additional clarification on RAI-TR101-SPLA-03 (external fires) and RAI-TR101-SPLA-06 (external flooding), which was provided in a letter dated February 8, 2008.

The applicant added consideration of external fires to the external events PRA and provided additional information on flooding caused by storm surge.

The methods used by the applicant to analyze external hazards, as documented in TR-101 and described in the DCD, are consistent with RG 1.200. Therefore, the external events analysis is acceptable to the staff given the input parameters used. There are two exceptions: the release of hazardous materials from nearby facilities and the treatment of high winds.

The analysis neither included an explicit discussion of the release of hazardous materials from nearby facilities (other than pipelines) nor identified this issue as a COL information item. The staff is concerned that some toxic materials are immediately dangerous to life and health at concentrations lower than the materials evaluated for pipelines, and some may not be readily detected. In RAI-SRP19.0-SPLA-17, the staff requested an assessment of risk from the release of toxic materials and a basis for a COL applicant to confirm that the assessment bounds the risk at the proposed site.

In a letter dated March 9, 2009, the applicant addressed the release of hazardous materials from nearby facilities and provided justification for screening of toxic releases from further analysis. The applicant stated that no operator action was credited, obviating the need to evaluate specific toxic release events with respect to type and amount of material released. The result of the analysis was a conditional core damage probability of 6.26×10^{-8}. From this, a limiting event frequency was provided for COL applicants to use to confirm that the generic analysis is applicable to their proposed sites. The applicant identified several conservatisms in this analysis. In addition to the assumption that operators were immediately and completely unable to perform any protective or mitigating actions, design features that assure control room habitability for 72 hours, under nearly all circumstances, were not credited.

In the same response, the applicant clarified the basis for using an initiating event frequency of 1×10^{-6} for the analysis of marine explosions. The applicant confirmed that screening of the event was based on a negligible contribution to core damage frequency so long as the criteria of RG 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," Revision 1 are met.

The staff agrees that there is considerable conservatism in the analysis of toxic gas release events that was described, and finds that it provides an acceptable basis for screening such events from further risk assessment. The limiting event frequency for toxic releases in the DCD provides an appropriate basis for COL applicants to confirm that this analysis bounds conditions where they propose to build a plant that references the AP1000 certified design. The staff considers the description of external events from transportation and nearby facility accidents to be complete and RAI-SRP19.0-SPLA-17 is resolved.

The applicant did not address the case of high winds while in MODE 5 and MODE 6. This scenario should be screened from consideration or the associated risks quantified and possibly controlled. The applicant addressed this concern in the March 26, 2009 response to RAI-SRP19.0-SPLA-18. The staff's evaluation is provided in Section 19.1.4.3.6 of this report.

The COL applicant must verify that the generic analysis for each external event bounds conditions at the proposed site. In a letter dated August 23, 2010, the applicant provided a COL information item to ensure that this action is taken by COL applicants.

DCD Section 2.2 requires a COL applicant to identify design changes in its safety analysis report if the occurrence of an external event that leads to severe consequences is 1×10^{-6} per year or greater. Accordingly, the COL applicant should assess the risk associated with any safety hazard that does not meet this criterion for being screened from further evaluation. In addition, the licensee must reevaluate the external event risk when a site-specific, plant-specific PRA is available.

The criteria for screening out external events from the quantitative evaluation are adjusted to maintain (for a plant referencing the AP1000 design) the conclusion reached in the DC—that the AP1000 design represents a reduction in risk compared to existing plants. The AP1000 DCD uses the following criteria with respect to risk evaluation of external events or hazards:

- An event or hazard with frequency less than 1×10^{-7} per year is screened from further evaluation.

- An event or hazard with frequency of 1×10^{-7} per year or higher is screened from further evaluation if a qualitative or bounding analysis shows that the associated CDF is less than 1×10^{-8} per year.

- An event or hazard with frequency of 1×10^{-7} per year or higher that cannot be shown to contribute less than 1×10^{-8} per year to CDF must be addressed in the risk analysis.

Each COL applicant must confirm that the high winds, floods, and other external events analysis documented in the DCD are applicable to the site for which the COL application is submitted (i.e., the spectrum of events at the site is bounded by the events analyzed in the DCD). Chapter 19 of the COL final safety analysis report (FSAR) should document this applicability evaluation. Further evaluation will be required if any unbounded, site-specific susceptibilities are found.

The NRC requires, where applicable to the site, that the COL applicant perform a site-specific, PRA-based analysis of external flooding, hurricanes, or other external events pertinent to the site to reveal any site-specific vulnerabilities. It is sufficient for the COL applicant to provide the basis for a conclusion that, for the proposed site, a particular external event is no more frequent and no more severe than that same event as modeled for the certified design. The COL licensee must develop plant-specific and site-specific risk information before loading fuel. This is part of COL Information Item 19.59.10-2.

In addition, the PRA used to support the AP1000 DC will be updated, as necessary, when site-specific and plant-specific (as-built) data become available. The COL applicant will review differences between the as-built plant and the design used as the basis for the AP1000 PRA to determine whether the PRA results are significantly impacted. Special emphasis should be placed on areas of the design that either were not part of the certified design or were not detailed in the certification. This is part of COL Information Item 19.59.10-2.

In RAI-SRP19.0-SPLA-02, the staff asked the applicant to clarify how SSCs are designed to withstand the effects of flooding. In its July 22, 2008, response letter, the applicant responded that the AP1000 is protected against floods up to the 30.5 meter (m) (100-foot (ft)) level. The 30.5 m (100-ft) level corresponds to the plant ground level. From this point, the ground is graded so that water will naturally flow away from the structures. Additionally, all seismic Category I SSCs below grade (below ground level) are designed to withstand hydrostatic pressures, and they are protected against flooding by a water barrier consisting of waterstops and a waterproofing system.

The staff finds that the design of safety-related SSCs below the 30.5 m (100-ft) level provides adequate protection from the effects of external flooding. Section 3.4 of this report discusses the staff's evaluation of internal flooding.

The COL applicant referencing the AP1000 certified design is responsible for: (1) confirming in the COL application that the information provided in Section 19.58 of the DCD is applicable to the selected site; and (2) addressing all site-specific action items discussed in Section 19.58 of the DCD.

The staff concluded that the methods used in the AP1000 PRA to evaluate external events provide the insights necessary to determine whether any design or procedural vulnerabilities exist for these external events. These methods provide insights needed for DC requirements, such as ITAAC. The staff finds that, for the events specified in the DCD, the reported results are acceptable. However, the case of high winds while in a shutdown mode had not been addressed. The staff identified this as Open Item OI-SRP19.0-SPLA-18. Subsequently, additional information was provided by the applicant; the staff's evaluation is documented in Section 19.1.4.3.2 of this report.

19.1.5.1 Probabilistic-Risk-Assessment–Based Seismic Margin Analysis

All seismic Category I SSCs are designed to remain functional when subjected to an earthquake, defined in 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants" as the safe shutdown earthquake (SSE). 10 CFR 100.23, "Geologic and seismic siting criteria," also applies to site-specific seismic hazard. The seismic analysis and design of the AP1000 plant is based on the certified seismic design response spectra (CSDRS) shown in DCD Tier 1, Figures 1.0-1 and 1.0-2. The CSDRS are based on RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," with an enhanced spectral acceleration in the 25-hertz (Hz) region. Its peak ground acceleration (PGA) is 0.3 g, while its dominant energy content is in the frequency range of 2 to 10 Hz.

In the DC amendment, the applicant presented another set of seismic response spectra in DCD Tier 1, Figures 1.0-3 and 1.0-4. The results of the applicant's analyses for the new set of seismic response spectra are shown in TR-115, APP-GW-GLR-115, "Effect of High-Frequency Seismic Content on SSCs," Revision 1, October 2008. This is a seismic analysis of the AP1000 nuclear island using hard-rock, high-frequency (HRHF) spectra that bound three different site conditions in the central and eastern United States. In addition to the results of these linear-elastic analyses for the design basis load determination, the applicant conducted detailed foundation structure interaction analyses using an approved coherency function to account for the scattering effect of seismic input. The effect of incoherency in seismic input at different points on a large foundation slab tends to reduce response for SSCs with natural frequencies from 25 to 50 Hz. TR-115 also provides supplemental criteria for selection and testing of equipment whose function might be sensitive to high-frequency acceleration. Section 3.7 of this report discusses the staff's evaluation of the seismic design.

In SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” Section II.N, “Site Specific Probabilistic Risk Assessments and Analysis of External Events,” the staff recommended a PRA-based seismic margin approach. At the plant level, high confidence in low probability of failure (HCLPF) should be established by ensuring that the seismic capacity of SSCs needed for safe shutdown is much larger than required for the design-basis earthquake. In a staff requirements memorandum dated July 21, 1993, the Commission modified that recommendation:

> PRA insights will be used to support a margins type assessment of seismic events. A PRA based seismic margins analysis will consider sequence level HCLPF [value]s and fragilities for all sequences leading to core damage or containment failures up to approximately one and two thirds the ground motion acceleration of the design-basis safe shutdown earthquake (SSE).

The applicant established a review-level earthquake (RLE) to demonstrate a one and two-thirds margin over the SSE, corresponding to the seismic design response spectra specified in DCD Tier 1, Figures 1.0-1, 1.0-2, 1.0-3 and 1.0-4.

For specific sites, the ground motion response spectra (GMRS) are obtained from site-specific, probabilistic seismic hazard–based analysis. Many of the GMRS of the central and eastern U.S. rock sites show response amplitudes that exceed the CSDRS for some frequency ranges. For this reason, an HRHF spectrum has been developed that bounds three hard-rock sites where AP1000 plants are proposed. DCD Figures 3I.1-1 and 3I.1-2 compare the HRHF at foundation level against the AP1000 CSDRS for both the horizontal and vertical directions for 5-percent damping. The HRHF spectrum exceeds the CSDRS for a range of frequencies above about 15 Hz. At high frequencies of vibratory excitation, the relative displacement is small and produces insignificant increase in stress. As an example, at 25 Hz and a spectral acceleration of 1.0g, the relative displacement is 0.016 inch. This is too small to cause damage.

In TR-115, the applicant evaluated representative SSCs that have been identified as “potentially sensitive to high-frequency input” in locations where the GMRS demonstrated an exceedance (magnitude greater than the CSDRS) in the high-frequency region.

In TR-144, APP-GW-GLN-144, “AP1000 Design Control Document High Frequency Seismic Tier 1 Changes,” Revision B, December 2007, the applicant stated that additional equipment dynamic qualification effort beyond the seismic design bases for operating nuclear power plants to address high-frequency response effects is not warranted. However, the applicant noted that the effect of high-frequency input on potentially sensitive active components requires additional consideration in accordance with interim staff guidance DC/COL-ISG-1.

In TR-144, the applicant concluded that, for structures, the HRHF loads would not govern the design. For the primary component supports and reactor coolant loop nozzles, seismic loads from the CSDRS bound those from the high-frequency input. Consequently, the staff considered these items to be acceptable seismic design for the HRHF input. For piping systems, the applicant concluded that the results of the HRHF seismic analysis are bounded by the stress results of the AP1000 CSDRS seismic analysis.

For safety-related electrical equipment, the applicant concluded that the qualification methodology (analytical evaluations and testing procedures) currently employed generally leads to a more conservative design than that resulting from the HRHF spectra. Supplemental seismic testing of high-frequency-sensitive safety-related equipment or implementation of one of the high-frequency screening techniques, approved in DC/COL-ISG-1, may be required to demonstrate acceptability under HRHF seismic demand conditions. Based on the acceptability of the seismic design and supplemental criteria for potentially susceptible equipment, the applicant stated that the conclusions of the PRA-based SMA are unchanged. In a recent update to seismic PRA based systems analysis, the applicant conducted a PRA sensitivity study ignoring the functions of all safety related electrical equipment; this study shows that the plant-level HCLPF value remains unchanged. This is primarily because of the passive design where key active components are designed to perform their safety functions when nonsafety-related support systems fail. This point is discussed further below.

Section 3.7 of this report discusses the staff evaluation of the seismic design. Subsequent to the application, the staff issued Interim Staff Guidance, DC/COL-ISG-20. This clarifies the staff's expectations for information to be included with an application for DC and the application for a license. It also identifies actions that the COL licensee must take to verify that the plant was built as designed.

For structures, primary component supports, and reactor coolant loop nozzles, as well as piping systems, the staff concludes that the applicant has demonstrated that, for the range of frequencies relevant to these SSCs, seismic loads are enveloped by CSDRS. This provides a sufficient basis for the staff to conclude that the seismic margins for these SSCs are acceptable.

In addition to the new parameters for the AP1000 generic site, the application for amendment to the certified design proposed a number of design changes that may affect the seismic capacity of equipment required to bring the plant to a safe, stable condition and to maintain containment integrity. Some proposed changes added seismic Category I SSCs to the design; others were deleted. These changes have the potential to affect the AP1000 SMA. Because of the changes, the staff requested an updated description of the results and insights of the AP1000 SMA.

Moreover, for safety-related equipment that is potentially sensitive to high-frequency excitation, the staff could not conclude that the applicant demonstrated adequate seismic margin, given the higher amplitude of high-frequency components of the GMRS. Although safety-related equipment that exhibits natural frequencies within the HRHF exceedance range will be subject to supplemental high-frequency seismic evaluation to confirm an acceptable seismic design, the applicant must clarify the basis for confirming that seismic margin is adequate.

The SMA for the certified design identifies HCLPF at the sequence level using a minimum-maximum[6] approach. Each COL applicant referencing the design should describe relevant site features and provide the basis for concluding that an acceptable seismic margin is maintained using this method or an alternative that is adequately justified. The staff identified these issues as Open Item OI-SRP19.0-SPLA-12.

In a letter dated August 23, 2010, the applicant provided a proposed revision to Section 19.55 of the DCD, "Seismic Margin Analysis," reporting the results of an updated, PRA-based SMA that

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[6] In the minimum-maximum or "min/max" approach, in a sequence where the failure of any individual SSC would cause core damage, the lowest individual SSC HCLPF value is used to characterize the entire sequence. If there is a sequence where the failure of multiple SSCs must occur to result in core damage, the highest HCLPF value for any of those SSCs is used.

reflects the current site parameters for the standard design. In Section 19.59.10.5, "Combined License Information," new and revised COL information items were proposed. The staff finds that the proposed changes to the DCD are consistent with DC/COL-ISG-3 and DC/COL-ISG-20; therefore, Open Item OI-SRP19.0-SPLA-12 is resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19.1.5.1.1 Dominant Accident Sequences for Seismic Events

The applicant's risk-informed SMA identified accident sequences for seismic events. For the SMA, the dominant sequences and associated cutsets are those that limit HCLPF values for the plant, irrespective of their likelihood. The margins approach does not support the determination that these are important contributors to seismic risk in a probabilistic sense, but allows identification of the plant features that are important to the plant level HCLPF value. The redundancy and diversity available in achieving that HCLPF value can also be confirmed.

The PRA-based SMA shows that the AP1000 design meets or exceeds the 0.5g HCLPF value. The applicant also performed a bounding analysis, using simplified and conservative assumptions, to identify paths by which the containment could be bypassed, fail to isolate, or fail. This analysis assumed that the reactor vessel fails because of failure of the fuel (HCLPF value 0.5g) and that the containment fails if the reactor vessel fails. Thus, the plant HCLPF for large release is the same as for core damage. This is an artifact of the applicant's conservative approach to SMA. For the AP1000, the conditional containment failure probability is nearly a full order of magnitude less than 1.0; LRF is much less than CDF. Because of design features to ensure in-vessel retention of the core, these are very conservative assumptions. Nevertheless, the AP1000 satisfies the expectation of the Commission as expressed in SECY-93-087 that the plant HCLPF value will be at least one and two-thirds times the SSE. Therefore, the staff concludes that seismic risk for the AP1000 design is acceptable.

COL applicants will confirm that site-specific features do not reduce the HCLPF value of a proposed plant. The capacity of as-built SSCs will be confirmed by a seismic walkdown to be performed after construction; the plant HCLPF value must remain at least as high as the value for the certified design.

The updated SMA was performed in a manner that was essentially identical to the SMA for the certified design. The capacity of those components required to bring the plant to a safe, stable condition was assessed. In some cases, for a particular SSC, a different method of calculating the seismic capacity of an SSC was selected from among several options that are acceptable to the staff.

In the risk-based SMA, no credit is taken for nonsafety-related systems. Because such systems are not seismic Category I, it is assumed that they become unavailable as a consequence of the seismic initiating event. The HCLPF value associated with transmission line ceramic insulators is low (0.09g), so all seismic events are assumed to entail loss of off-site power. Since the diesel generators are nonsafety-related and, therefore, assumed not to be available, all seismic accident sequences involve station blackout (loss of all alternating current (ac) power). The analysis investigated and accounted for the potential for adverse interactions between nonsafety-related SSCs (assumed to be damaged) and safety-related systems. The event and fault trees developed for the internal events PRA were modified to accommodate seismic events. In this way, the random failures and human errors modeled in the internal events portion of the PRA are captured in the seismic analysis.

The modified event and fault trees were merged and cutsets for all sequences that lead to core damage were generated. Most of the HCLPF values for components and structures were obtained through computing a conservative deterministic failure margin (CDFM), performing a probabilistic fragility analysis, or using deterministic methods. The HCLPF values for electrical equipment, where test results are documented, were obtained by comparing required response spectra to test response spectra for similar types of equipment.

19.1.5.1.2 Risk-important Features and Operator Actions for Seismic Events

The limiting SSCs for which seismically induced failure would lead directly to core damage include the pressurizer and the fuel in the reactor vessel (a HCLPF value of 0.5 g). This HCLPF value was also assumed for a large number of SSCs that may be sensitive to high frequency excitation. By design, even if these frequency-sensitive components are determined to have a lower seismic capacity, the plant HCLPF value would remain the same (0.5 g).

Updated HCLPF values for most structures indicate higher seismic capacity. An exception is the polar crane: due to a change in the design loading of the crane, it has become the limiting SSC for gross structural failure (0.55 g). Although parts of nonsafety-related structures have been upgraded to seismic Category II, the SMA did not take credit for these changes. The staff finds this to be conservative and therefore acceptable.

Operator actions are not credited in the SMA model for events that control the plant HCLPF value. The staff concludes that this is conservative, as the inclusion of operator actions in the models would only have the potential to increase the plant HCLPF.

19.1.5.1.3 Insights from Uncertainty, Importance, and Sensitivity Analyses for Seismic Events

Uncertainty analysis for seismic events is performed to quantify the range of values within which the results of an analysis could reasonably be expected to fall. Hazard curves, the result of a seismic hazards analysis, have a large uncertainty due to variability in source term and attenuation relationships. The large uncertainties in the hazard curves will dominate the resulting core damage frequency analysis, but uncertainties in equipment and structure fragilities are subject to much lower variability, since more and better information exist for them. This makes sensitivity analysis for SSCs less meaningful. Consequently, staff finds that performing a convolution of seismic hazard and fragility is not necessary for a generic plant design; consequently, a sensitivity analysis of plant HCLPF to SSC fragility is not meaningful. No additional uncertainty analysis was performed because uncertainty is directly addressed in the margins method. HCLPF values represent the seismic capacity (peak acceleration expressed in terms of the acceleration of gravity) at which there is 95 percent confidence that equipment needed for safe shutdown will fail less than 5 percent of the time. This provides margin to bound uncertainty while avoiding the extremes of the probability distributions.

Because the margins method does not quantify risk, importance analyses were not performed. The applicant did, however, perform an additional sensitivity analysis on the effects of changes in certain assumptions used in the SMA. The applicant chose to vary the HCLPF values for equipment that is to be seismically qualified by testing. When these values are varied from 0.5 g to 0.3 g, the HCLPF value is not changed for any sequence or event. Consequently, the plant HCLPF value was also unchanged.

The AP1000 SMA has shown that the plant HCLPF value is at least one and two-thirds the ground motion acceleration of the design-basis SSE. Because it limits the plant HCLPF value (0.50 g), the nuclear fuel is included within the scope of the reliability assurance program (RAP). The pressurizer, although its upper support weld was assigned the same HCLPF value in the SMA, was not included in the RAP because an alternative method of analysis shows it to have a significantly higher HCLPF value (0.56 g, based on the upper support strut). Because the SMA takes no credit for nonsafety-related systems to mitigate seismic events, the results of the SMA do not affect the probabilistic criteria used to select nonsafety-related SSCs for inclusion in the RAP. The staff concludes that the result of PRA-based SMA, as amended to include hard-rock and soil sites, meets the Commission expectation for adequate margin expressed in its staff requirements memorandum (SRM) to SECY-93-087.

19.1.5.4 High Winds Evaluation

High winds can affect plant structures in two ways: (1) structures can collapse or overturn from the excessive loading when wind forces exceed the load capacity of the structure; and (2) lifting and thrusting can cause materials to act as missiles against plant structures that house safety-related equipment. In addition, the applicant investigated the potential for debris generated by high winds to clog the passive containment cooling system (PCS) drains and directly or indirectly to block the PCS air baffle.

The AP1000 structures protecting safety-related features are designed to withstand winds of up to 483 kilometers per hour (km/h) per hour (300 miles per hour (mph)), as well as missiles generated by these winds (see design-basis wind speed discussed in Chapter 2 of the DCD). Also, the AP1000 operating basis wind speed is 233 km/h (145 mph), as discussed in Chapter 2 of the DCD. In general, there is some margin above the design and operating bases that the risk evaluation of high winds neither assesses nor credits. The applicant made the following assumptions in evaluating the risk from high winds:

- Safety-related structures, which house safety-related equipment, are not impacted by high winds of any kind (tornados and hurricanes, including extra-tropical cyclones) if the wind speed does not exceed 483 km/h (300 mph).
- Nonsafety-related structures, which are designed and built according to uniform building code and house nonsafety-related defense-in-depth or investment protection equipment, are not impacted by high winds of any kind (tornados and hurricanes, including extra-tropical cyclones) if the wind speed does not exceed 233 km/h (145 mph).
- High-wind events exceeding 483 km/h (300 mph) are extremely rare events with a frequency of less than $1{\times}10^{-7}$ per year; therefore, they are screened out from the risk analysis based on the screening criteria discussed in Section 19.1.5, above. The COL applicant referencing the AP1000 design must verify this assumption.

The applicant states in DCD Section 19.58.2.1 that no tornados or hurricanes are expected to reach 483 km/h (300 mph) winds per the enhanced Fujita scale for tornados and the Saffir-Simpson scale for hurricanes. Though the staff does not assign an upper wind speed limit to these scales, the conclusion is consistent with the staff's position documented in RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1. For the continental United States, the staff considers the highest tornado wind speed with a frequency of $1{\times}10^{-7}$ to be 370 km/h (230 mph). AP1000 safety-related structures are designed to withstand winds of 483 km/h (300 mph). Clearly, the expected frequency of 483 km/h

(300 mph) tornadoes is significantly lower. For plants that are to be sited in the continental U.S., such events may be screened from further analysis.

Based on these assumptions, the applicant performed generic risk evaluations using initiating event frequencies that it described as "bounding." Six tornado event categories (defined in Table 19.58-1 of the DCD, which describes the enhanced Fujita scale for tornados) and five hurricane event categories (defined in Table 19.58-2 of the DCD, which describes the Saffir-Simpson scale for hurricanes) were evaluated. In addition, the applicant considered extra-tropical cyclones as a single category of high winds. Extra-tropical cyclones are normal storms and thunderstorms with winds expected to fall below the operating basis of 233 km/h (145 mph). The analysis assumed a bounding frequency of extra-tropical cyclones equal to 3×10^{-2} per year. COL applicants referencing the AP1000 design must verify that the frequency of each of the 12 high wind categories at the proposed site is bounded by the frequency assumed in Section 19.58 of the AP1000 DCD.

High winds cannot be screened out from the evaluation using the initiating event frequency criterion because the assumed frequency of high winds is greater than 1×10^{-7} per year. However, bounding risk assessments have shown that the CDF associated with high winds is less than the criterion of 1×10^{-8} per year. Therefore, the applicant did not perform detailed risk assessments for high winds. The applicant did perform risk assessments for three cases—a baseline case and two sensitivity cases:

- The baseline case assumes two kinds of failures: (1) an unrecoverable LOSP event for all 12 high wind categories since the site switchyard is unprotected; and (2) failure of all nonsafety-related structures for three categories of tornados (designated as EF3, EF4, and EF5 in Table 19.58-1 of the DCD) and three categories of hurricanes (designated as Category 3, 4, and 5 in Table 19.58-2 of the DCD), which exceed the operating basis of 233-km/h (145-mph) winds that could impact nonsafety-related structures. The applicant assumed that the failure of the nonsafety-related structures leads to the failure of all nonsafety-related systems credited in the AP1000 PRA with the exception of the manual DAS. The DAS manual actuation cables are located within the nuclear island and are, therefore, protected against high winds.

- The first sensitivity case assumes an unrecoverable LOSP event for all 12 high wind categories but no other failures. This sensitivity case removes the conservative assumption that all nonsafety-related structures fail when the operating basis of 233 km/h (145 mph) for high winds is exceeded, even though all structures are designed with some margin to withstand winds above the operating basis.

- The second sensitivity case assumes that all 12 high wind categories cause both an unrecoverable LOSP event and the failure of all nonsafety-related structures. This sensitivity case is a very conservative upper case since it assumes that all nonsafety-related structures fail even for categories of high winds that do not exceed the operating basis of 233 km/h (145 mph).

Table 19.58-3 of the AP1000 DCD summarizes the three risk assessment cases. The estimated CDF for the baseline case is about 5×10^{-9} per year, which is less than the criterion of 1×10^{-8} per year. Therefore, no detailed risk assessment of high winds is necessary. The CDF for the first and second sensitivity cases are about 2.3×10^{-9} and 1.4×10^{-8}, respectively. The first sensitivity case indicates that the estimated risk is not significantly sensitive to assumptions about the impact on nonsafety-related structures of high winds exceeding the operating basis.

The second sensitivity case indicates that the screening criterion of CDF less than 1×10^{-8} per year is almost met even under very conservative assumptions about the failure of nonsafety-related structures. The staff finds that the bounding risk assessments documented in Section 19.58 of the AP1000 DCD show that, under the stated assumptions for external events (which must be verified by the COL applicant), the risk from high winds (tornados and hurricanes) is so small that no detailed risk assessments for high winds are needed. In RAI-SRP19.0-SPLA-03, the staff requested the addition of COL information items to enumerate the assumptions to be verified. In a letter dated August 21, 2008, the applicant proposed to amend the DCD to state, "A site specific review of the generic PRA should be conducted to verify that the assumptions in the PRA bound the site specific conditions for the applicant's site." In a letter dated August 23, 2010, the applicant proposed a revision to Section 19.59.10.5, "Combined License Information," with a revised COL information item to make this explicit. The staff finds that the proposed change is consistent with RG 1.200 and acceptable; therefore, RAI-SRP19.0-SPLA-03 is resolved.

In addition to structural failures, the applicant investigated the potential for blockage or plugging of the PCS airflow path by debris generated by high winds. The applicant found that blockage or plugging of the PCS airflow path could occur by blockage of the screens, failure of the louvers, or blockage of the chimney outlet. However, because of the presence of certain design features and operational requirements, these failure mechanisms are highly unlikely (i.e., their frequency is smaller than 1×10^{-7} per year). For this reason, the PRA does not model them:

- Large openings are distributed around the circumference of the shield building, just underneath the roof. Screens and fixed, always-open louvers designed to prevent foreign objects or debris from entering the air flowpath cover these openings, providing an area through which air can flow into an enclosed plenum.

- Ducts made of pipe slant upward from this plenum through the shield wall. They supply air to a common plenum above the outer flow annulus inside the shield building.

- A walkway provides access for inspection of the flowpaths and removal of debris.

- There is a surveillance requirement to verify that the airflow path is unobstructed.

- The chimney outlet is designed to produce the necessary airflow in the event of an accident. The outlet contains heavy grates to guard against missiles. Screens prevent the entry of foreign objects into the annulus around containment. During normal operation, a positive airflow prevents ice and snow from entering the chimney.

In RAI-SRP19.0-SPLA-01, the staff requested that insights related to high winds and containment cooling be added to the DCD (RAI-SRP19.0-SPLA-01 also discussed external flooding as addressed in Section 19.1.5.5). In a letter dated July 22, 2008, the applicant proposed a revision to DCD Table 19.59-18, identifying these features and requirements among the PRA assumptions and insights. This is consistent with NUREG-0800 and is, therefore, acceptable to the staff.

In a letter dated March 4, 2008, the applicant responded to a request for information from the staff (RAI-TR142-SPCV-02 through RAI-TR142-SPCV-04). The applicant provided the results of an analysis demonstrating that most of the inlet area would have to be blocked before the PCS function is degraded. In addition, the applicant stated the following:

> There is a possibility that these [inlet] screens could become clogged with airborne debris. For this reason, there is access to the louvers and screens by an enclosed walkway between the wall containing the louvers and the shield building wall. Regular inspections of the louvers will be made and the screens will be kept free of debris. The frequency of inspections is expected to be once per month, and may change depending on the degree of blockage observed.

The staff finds that this commitment and the surveillance requirement (SR) provide assurance that significant air blockages will not exist before high winds or ice storms. Because of the very large degree of blockage that would be required to challenge containment cooling, the elevation of the openings, and the design of the airflow path, the staff considers the frequency of such an event to be negligible.

In DCD Revision 17, insights related to the protected location of DAS manual actuation cables and the assurance of adequate containment cooling air flow have been added to Table 19.59-18. The staff considers the portions of RAI-SRP19.0-SPLA-01 related to high winds to be resolved.

19.1.5.5 External Flooding Evaluation

The applicant assessed various scenarios with the potential to raise water levels and concluded that, even in combination, external flooding of safety-related SSCs (and certain risk-significant investment protection equipment) is prevented. For that reason, the effect of external flooding on risk-significant SSCs was not evaluated. Although storms, dam failure, and other external phenomena can cause flooding, the analysis assumed that the generic site was not subject to flooding by flash floods or failure of an upstream dam. COL applicants must show that this is also true for their proposed site. Otherwise, COL applicants must evaluate the other external events that can raise water levels at the site.

The risk evaluation of external floods is based on the following features from the design basis of the plant documented in Section 2 of the AP1000 DCD:

- The AP1000 is protected against floods up to the 30.5 m (100-ft) level, which corresponds to the plant ground level. From this point, the ground is graded so that water naturally flows away from the plant structures.

- The plant is designed such that the 30.5 m (100-ft) level is slightly above grade and the level of anticipated external flooding. Below grade is protected against flooding by a water barrier consisting of waterstops and a waterproofing system. Seismic Category I SSCs below grade are designed to withstand hydrostatic pressures.

- The seismic Category I SSCs below grade (below ground level) are protected against flooding by a water barrier consisting of waterstops and a waterproofing system.

In RAI-SRP19.0-SPLA-01, the staff also requested that these insights related to external flooding be added to the DCD (RAI-SRP19.0-SPLA-01 also discussed high winds cooling, addressed in Section 19.1.5.4).

In a letter dated July 22, 2008, the applicant proposed a revision to DCD Table 19.59-18 identifying these features and requirements among the PRA assumptions and insights. This is consistent with NUREG-0800 and is, therefore, acceptable to the staff. In DCD Revision 17, insights related to external flooding have been added to Table 19.59-18. The staff considers the portions of RAI-SRP19.0-SPLA-01 related to external flooding to be resolved.

The risk evaluation of external floods is highly specific to the proposed plant site. A detailed risk analysis of external floods requires information not only on potential sources of water but also on local configurations such as dikes, surface grading, locations of structures, and location of equipment within the structures. However, bounding assumptions about candidate sites are used to show that external floods can be screened from detailed risk evaluation. Assuming that the ground is graded away from the structures and there is no site susceptibility to dam failure or flash flooding, the remaining source of external flooding is storm surges capable of reaching the plant ground level. Hurricanes can produce the highest storm surges. The applicant performed a screening risk evaluation using as a reference the site most susceptible to external floods from hurricane surge water among potential candidate sites for an AP1000 plant. This site is located at an elevation of 13.7 m (45 ft) above sea level and in an area where the highest storm surges due to hurricanes have occurred. All other proposed sites are located at higher elevations above sea level. Therefore, it would require a 13.7 m (45-ft) hurricane storm surge to reach the plant ground level. Any surge that stops below ground level at the plant has no impact on the plant due to flooding. Based on the Saffir-Simpson hurricane scale (Table 19.58-2 of the AP1000 DCD), only Category 5 hurricanes have the ability to generate storm surges in excess of 5.49 m (18 ft). Historically, the highest observed storm surges occurred during hurricane Katrina in 2005 and hurricane Camille in 1969. The maximum high water mark observation occurred during hurricane Katrina with a surge of 8.47 m (27.8 ft) above normal tide levels at Pass Christian, on the immediate Gulf Coast just east of St. Louis Bay.

Based on the historical information, documented in Section 19.58.2.2 of the DCD, the applicant stated that a hurricane storm surge in excess of 8.53 m (28 ft) can be classified as a rare event and a hurricane storm surge in excess of 13.7 m (45 ft) as an extremely rare event and can be assigned a frequency of 1×10^{-7} per year or less. In addition, a risk assessment that was performed as a sensitivity study, which assumed loss of the switchyard and all nonsafety-related SSCs, indicated that the CDF associated with external floods is insignificant when areas containing safety-related equipment are protected. Therefore, by recognizing the fact that the AP1000 design provides features (e.g., barriers) that provide protection against the propagation of flooding to areas where safety-related equipment is located, external floods that do not closely approach ground level at the plant are screened from detailed risk evaluation in accordance with the criteria discussed in Section 19.1.5.

On the basis of the discussion in DCD Section 19.58 and the large margin between the greatest storm surge observed and the water level required to affect plant safety, the staff finds that the frequency of external flooding caused by storm surge is negligibly small. The screening criteria are met and no further analysis is required. The staff expects a COL applicant to verify the applicability of the screening criteria to the proposed site.

COL applicants must confirm that all possible mechanisms of external flooding at the proposed site have been assessed, including credible combinations of those mechanisms. Otherwise, the COL applicant must screen out these external events by demonstrating that they occur with negligible frequency. COL Information Item 19.59.10-2 requires the COL applicant to re-evaluate the qualitative screening of external events.

Because this approach is consistent with RG 1.200, the staff considers this an acceptable way to address the risk of external flooding.

19.1.5.6 Transportation and Nearby Facilities Accident Evaluation

Section 19.58.2.3 of the AP1000 DCD discusses the risk from external events related to transportation accidents near the nuclear plant and to accidents at nearby industrial and military facilities. DCD Section 19.58 discusses the following types of accidents: (1) aviation, (2) marine, (3) pipeline, and (4) railroad and truck. The staff finds that these accidents form an acceptable set of transportation and nearby facility accidents based on the modified IPEEE guidelines discussed in Section 19.1.5, above. Each COL applicant should verify that these analyses bound all such hazards relevant to the proposed site.

19.1.5.6.1 Aviation Accidents

The risk evaluation for aviation accidents considers two cases: (1) small aircraft impact; and (2) commercial aircraft impact. The applicant screened small aircraft impact accidents from detailed risk evaluation by performing a bounding risk evaluation based on a limiting frequency of 1.2×10^{-6} impact events per year.

For small aircraft impact accidents, the applicant performed a bounding risk evaluation that assumed a limiting initiating event frequency of 1.2×10^{-6} per year, together with an LOSP and loss of nonsafety systems. The likelihood that the impact of a small aircraft would challenge the safety systems (all located within the nuclear island) is considered negligible. Assuming that a small aircraft impact could cause an LOSP and loss of nonsafety systems, the applicant showed that the associated CDF is less than 1×10^{-8} per year.

The applicant screened commercial-size aircraft impact accidents from detailed risk evaluation by assuming a limiting frequency of 1×10^{-7} impact events per year. Each COL applicant will demonstrate the assumed limiting event frequency for the selected site of 1.2×10^{-6} per year for small aircraft and 1×10^{-7} per year for commercial-size aircraft.

19.1.5.6.2 Marine Accidents

Sites close to large waterways with ship and/or barge traffic need to evaluate the risk associated with marine accidents. Marine accidents pose a hazard to a nuclear power plant due to: (1) release of hazardous material towards the plant; and (2) explosion with resulting damage to the plant.

The applicant evaluated the risk associated with the release of hazardous material towards the plant, following a marine accident, using a bounding analysis that assumed a limiting initiating event frequency of 1×10^{-6} per year, and showed that the associated CDF is less than 1×10^{-8} per year. Therefore, based on the screening criteria discussed in Section 19.1.5, the release of hazardous material in a marine accident requires no detailed risk evaluation. The applicant modeled the risk impact of a toxic release by assuming a reactor trip and guaranteed failure of all operator actions credited in the PRA (the toxic release is not expected to lead to any direct failure of safety equipment). This is a conservative analysis because the AP1000 has an additional level of defense against toxic airborne material. Specifically, with warning that a release has occurred, the operators can actuate passive control room habitability. This system isolates the control room from normal heating, ventilation, and air conditioning (HVAC) and actuates a separate system supplied from compressed air containers. The compressed air slightly pressurizes the control room above atmospheric pressure, preventing the entrance of toxic material for at least 72 hours. (This is adequate time for operators to deal with the event.)

The staff review finds that the bounding evaluation, documented in Section 19.58.2.3.2 of the AP1000 DCD, demonstrates that the risk associated with the release of hazardous materials in a marine accident is insignificant, assuming a limiting initiating event frequency of 1×10^{-6} per year. The COL applicant for the selected site will demonstrate the assumed frequency of 1×10^{-6} per year for release of hazardous materials that could pose a hazard to the plant by a marine accident.

The applicant screened out qualitatively the risk associated with an explosion following a marine accident with resulting damage to the plant from a detailed analysis based on the acceptance criteria of event frequency less than 1×10^{-7} per year and CDF less than 1×10^{-8} per year for the following reasons:

- Loss of service water events resulting from a marine explosion is not a nuclear safety concern for AP1000 since the design does not include a service water intake structure.

- RG 1.91 provides the acceptance criterion of an overpressure event in excess of 6.9 kilopascal (kPa) (1 pound per square inch (psi)) at a frequency of less than 1×10^{-6} per year.

- Margin above the RG 1.91 acceptance criterion has been demonstrated. A study for the Waterford site, NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States: Other External Events," Supplement 2, indicated that the AP1000 safety-related buildings can withstand overpressures above the RG 1.91 acceptance criterion of 6.9 kPa (1 psi).

The staff review finds that the risk associated with an explosion following a marine accident is insignificant when the RG 1.91 acceptance criterion of an overpressure event in excess of 1 psi at a frequency of less than 1×10^{-6} per year is met. The COL applicant will demonstrate that the RG 1.91 criterion of an overpressure event in excess of 1 psi at a frequency of less than 1×10^{-6} per year is met for the selected site.

19.1.5.6.3 Pipeline Accidents

Sites close to pipelines need to evaluate the risk associated with pipeline accidents. Pipeline accidents pose a hazard to a nuclear power plant because of the potential for: (1) a release of hazardous material towards the plant; and (2) an explosion with resulting damage to the plant. The applicant evaluated the risk associated with the release of hazardous material towards the plant following a pipeline accident using a bounding analysis that assumed a limiting initiating event frequency of 1×10^{-6} per year. This analysis showed that the associated CDF is less than 1×10^{-8} per year. Therefore, based on the screening criteria discussed in Section 19.1.5, the release of hazardous material in a pipeline accident requires no detailed risk evaluation. The applicant modeled the risk impact of a toxic release by assuming a reactor trip and guaranteed failure of all operator actions credited in the PRA (the toxic release is not expected to lead to any direct failure of safety equipment). This is a conservative analysis because the AP1000 has an additional level of defense against toxic airborne material. Specifically, with a warning that a release has occurred, the operators can actuate passive control room habitability. This system isolates the control room from normal HVAC and actuates a separate system supplied from compressed air containers. The compressed air slightly pressurizes the control room above atmospheric pressure, preventing the entrance of toxic material for 72 hours. (This is adequate time for operators to deal with the event.)

The risk associated with an explosion following a pipeline accident with resulting damage to the plant was screened out qualitatively from a detailed analysis, based on the acceptance criteria of event frequency less than 1×10^{-7} per year. Section 19.58.2.3.3 of the AP1000 DCD documents an approach (pipeline accident model) that qualitatively illustrates potential scenarios from gas pipeline accidents. This approach briefly discusses the following considerations for evaluating the frequency of pipeline accidents: (1) gas pipe rupture frequency estimation; (2) gas cloud formation probability estimation; (3) gas cloud transportation and nondispersion probability estimation; and (4) onsite gas cloud ignition probability estimation.

The staff review finds that the risk associated with pipeline accidents is insignificant, assuming that the COL applicant will demonstrate that the frequency criterion of 1×10^{-7} per year is met for the proposed site for pipeline accidents that could pose a hazard to the plant. The frequency of pipeline accidents will be evaluated by the COL applicant using the approach discussed in Section 19.58.2.3.3 of the DCD or another approach acceptable to the staff.

19.1.5.6.4 Railroad and Truck Accidents

Railroad and truck accidents could pose a hazard to an AP1000 plant, and COL applicants need to evaluate the risk associated with such accidents. As for marine and pipeline accidents, railroad and truck accidents could pose a hazard to a nuclear power plant because of the potential for: (1) a release of hazardous material towards the plant; and (2) an explosion with resulting damage to the plant. However, railroad and truck accidents are expected to be less likely to occur (e.g., because of the improved security barriers established at U.S. nuclear power plants) and cause less plant damage than aviation or marine accidents if they should happen. For these reasons, the risk impact from railroad and truck accidents is insignificant if the initiating event frequency criterion of 1×10^{-7} per year is met.

The staff review finds that the risk associated with railroad and truck accidents is insignificant, assuming that the COL applicant can demonstrate that the frequency criterion of 1×10^{-7} per year is met for the proposed site for railroad and truck accidents that could pose a hazard to the plant.

19.1.5.7 External Fires

External fires are those that occur outside the controlled site boundary. Potential effects on the plant could be LOSP, forced isolation of the plant ventilation, and control room evacuation. External fires are not expected to spread on site because of site clearing during the construction phase and control of combustibles during construction and operation. In RAI-TR101SPLA-03, the staff requested that the applicant consider external fires more explicitly.

In a letter dated February 8, 2008, the applicant agreed to address that based on site-specific information, the COL applicant should reevaluate the qualitative screening of external fires. Accordingly, based on the criteria discussed in Section 19.1.5, above, which were used to screen out external hazards in the PRA, a risk evaluation should be performed if the COL applicant cannot demonstrate that the frequency of external fires that could pose a hazard to the plant is less than 1×10^{-7} per year. If the COL applicant identifies any site-specific susceptibilities, the site-specific PRA performed by licensees to address COL Information Item 19.59.10-2 should include external fires.

This is consistent with RG 1.200 and, therefore, acceptable to the staff. The staff considers RAI-TR101SPLA-03 resolved.

19.1.5.8 Conclusions

Information documented in Section 19.58 of the AP1000 DCD addresses the second part of COL Information Item 19.59.10-2 which reads as follows:

> Based on site-specific information, the COL should also re-evaluate the qualitative screening of external events…. If any site-specific susceptibilities are found, the PRA should be updated to include the applicable external event.

The information provided in Section 19.58 of the AP1000 DCD includes the following objectives:

- to show that screening criteria are met and to identify external events that may impact the AP1000 risk on a site-specific basis

- to provide generic risk analyses, based on bounding assumptions regarding site-specific parameters (e.g., frequency of each category of hurricanes) for some external events

- to provide guidance to COL applicants regarding the verification of the applicability of these “generic” analyses to a specific site

Based on modified IPEEE guidelines, DCD Section 19.58 discusses the following external events or external hazards:

- high winds (including tornadoes)
- external floods
- transportation and nearby facility accidents
- external fires

The staff review finds that these external hazards are most likely a complete list of events associated with candidate sites for an AP1000 plant. However, as stated in DCD Section 2.2.1, “Combined License Information for Identification of Site-specific Potential Hazards,” the COL applicant must verify that this list adequately addresses external hazards at the proposed site. The COL applicant should use site-specific information to verify that the assumptions made in the analyses performed during the DC stage are applicable. For example, screening on the basis of event frequency of external flooding due to tsunamis or upstream dam failures may not be possible at all sites.

The staff finds that the generic risk analyses and other information provided in Section 19.58 of the AP1000 DCD are acceptable, including the screening of events from inclusion in the PRA, given the documented assumptions.

However, these analyses are based on assumptions that are expected to envelop site-specific information at sites selected to build a nuclear plant referencing the AP1000 design. The COL applicant referencing the information provided in DCD Section 19.58 must: (1) confirm in the COL application that the information provided in Section 19.58 of the DCD is applicable to the selected site; and (2) ensure that the assumptions made in the generic risk evaluations documented in Section 19.58 of the DCD bound the site-specific conditions for the applicant’s site. This is in agreement with the stipulation made in Section 19.58.3 of the AP1000 DCD, which states that the COL applicant should conduct a site-specific review of the generic PRA to verify that the assumptions in the PRA bound the site-specific conditions for the applicant's site (COL Information Item 19.59.10-2).

19.1.8.24 Reactor Pressure Vessel Thermal Insulation System

The AP1000 design includes a reflective RVIS that provides an engineered flow path to allow water ingress and venting of steam for external reactor vessel cooling (ERVC) in the event of a severe accident involving core relocation to the lower plenum. COL Action Item 19.2.3.3.1.3.2-1 calls for the COL applicants to complete the design for the RPV thermal insulation system. Section 39.10.2 of the AP1000 PRA specifies its functional requirements. In addition to RCS depressurization and reactor cavity flooding, several other conditions are necessary: (1) the reactor vessel thermal insulation system design must be consistent with ULPU Configuration V testing with prototypical insulation (ULPU-2000 is a boiling heat transfer test facility at the University of California at Santa Barbara used to investigate in-vessel retention of a damaged core); (2) the RVIS must maintain its integrity under the hydrodynamic loads associated with ERVC and not be subject to clogging of the coolant flow path by debris; and (3) RPV exterior coatings do not preclude the wetting phenomena identified as the cooling mechanism in the ULPU testing.

The applicant has completed the design of the RVIS. In TR-24, the applicant provided information to demonstrate that the RVIS is designed to provide adequate cooling to ensure in-vessel retention of a damaged and relocated core. On this basis, the applicant proposed to close COL Information Item 5.3-5. The staff's evaluation is in Section 19.2.3.3.1.3.2 of this report.

19.1.9 Conclusions and Findings

The staff has evaluated the AP1000 design PRA quality and its use in the design and certification processes. The NRC concludes that the quality and completeness of the AP1000 PRA are adequate for its intended purposes, which are to support the design and certification processes and satisfy the requirements of 10 CFR 52.47. The approaches used by the applicant for both the core damage and containment analyses are logical and sufficient to achieve the desired goals of describing and quantifying potential core damage scenarios and containment performance during severe accidents.

The use of PRA in the AP1000 design process improved the unique passive features of the design by providing a better understanding of plant response, including potential system interactions, during postulated accidents beyond the design basis. Such features contributed to the reduced CDF and conditional containment failure probability estimates of the AP1000 design when compared to those of operating PWRs. The applicant used the PRA results and insights to identify areas in which it is particularly important to implement the certification and operational requirements assumed during the design and certification processes (e.g., ITAAC, RTNSS requirements, D-RAP, COL information items, and technical specifications).

The staff reviewed the description and results of the PRA provided in the amendment. The staff finds that they are consistent with the requirements of 10 CFR 52.47 and the guidance provided in RG 1.200, NUREG-0800 Sections 19.0 and 19.1, as well as DC/COL-ISG-1, DC/COL-ISG-3 and DC/COL-ISG-20. Therefore, the staff concludes that the amended AP1000 design meets the NRC's safety goals and represents an improvement in safety over current operating PWRs in the United States.

19.1.10 Resolution of Safety Evaluation Report Open Items

Open Item OI-SRP19.0-SPLA-07: The applicant must implement corrective actions after the audit, re-solving and requantifying the corrected model as well as revising TR-102 and making associated changes to the DCD consistent with DC/COL-ISG-3.

As discussed in Section 19.1.2.1.10, "Improved Control Room Design and Digital Instrumentation and Control Systems," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19.0-SPLA-12: The applicant must confirm that an acceptable seismic margin is maintained for HRHF sites.

As discussed in Section 19.1.5, "Seismic Margin Analysis," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19.0-SPLA-13: The applicant must provide an updated DCD description of events (HAs, common cause, and basic) and sequences contributing most to risk, both at power and while shutdown.

As discussed in Section 19.1.4, "Safety Insights from the Internal Events Risk Analysis for Shutdown Operation," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19.0-SPLA-14: The applicant must resolve the discrepancy in the containment inventory of radionuclides used for the survivability evaluation, determining whether the environmental assessment should include mechanical penetrations and hatches (e.g., gasket materials) and providing a licensee COL information item to finalize the list of equipment that must survive.

As discussed in Section 19.2.3.3.7, "Equipment Survivability," this open item has been resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Open Item OI-SRP19F-SPLA-01: In a letter dated March 19, 2010, the applicant submitted the results of an analysis consistent with guidance from the Nuclear Energy Institute. It was prepared using NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," which has been endorsed in draft guide (DG)-1176, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts." The staff's evaluation of this aircraft impact assessment (AIA) is in Section 19F of this report.

19.1.11 Combined License Information Items

19.1.11.1 As-Built Seismic Margin Assessment

COL Information Item 19.59.10-1 (COL Action Items 19A.2-1 and 19A.2-2) is associated with an as-built SSC HCLPF comparison to seismic margin evaluation. In TR-6, the applicant noted that a COL applicant cannot complete the review and proposes instead to complete these actions before fuel loading.

However, at the time of COL application, site-specific features with the potential to affect the SMA should be described and assessed to confirm that the SMA documented in the DCD is applicable to a plant located on the proposed site. Alternatively, additional analysis to demonstrate adequate seismic margin is required. COL Information Item 19.5.10-6 is a COL applicant item.

Because the SMA requires a confirmatory walkdown, which cannot be performed until construction of seismic SSCs has been completed, the staff agrees that evaluation of as-built conditions cannot be provided with the COL application. The staff concludes that performance of the as-built SMA before fuel loading is timely and, therefore, acceptable. COL Information Item 19.59.10-1 should be completed by the licensee.

19.1.11.2 Site-Specific, Plant-Specific Probabilistic Risk Assessment

COL Information Item 19.59.10-2 is associated with evaluating an as-built plant versus design in AP1000 PRA and site-specific PRA external events. In TR-6, the applicant noted that a COL applicant cannot complete an as-built review and proposes that the COL applicant referencing the AP1000 certified design will, instead, review differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 19.59-18. The applicant proposed that, if a screening analysis shows that the effect of the differences could result in a significant increase in CDF or LRF, the PRA would be updated to reflect these differences. In addition, the COL applicant should reevaluate the qualitative screening of external events. If any site-specific susceptibilities are found, the PRA should be updated to include the applicable external event.

The staff agrees that the design-specific PRA is a sufficient and acceptable basis for drawing safety conclusions for a license and should be described in the COL FSAR (with appropriate discussion of plant-specific features and departures from the certified design). The FSAR should also identify key assumptions and insights from this PRA. The staff finds that documentation and qualitative screening of external events, specific to the proposed plant site, is an acceptable way to confirm that site-specific vulnerabilities do not require further risk assessment and need not be included in PRA results and insights reported in the FSAR. This part of COL Information Item 19.5.10-2 is a COL applicant item.

However, the staff finds that each licensee’s PRA should model significant plant-specific and site-specific differences from the design PRA, whether positive or negative, to be consistent with DC/COL-ISG-3. This is necessary to support operational-phase reliability assurance activities, when more realistic assessment of risk is needed to avoid masking activities of risk significance. This part of COL Information Item 19.5.10-2 is to be completed by the COL licensee.

19.2 Severe Accident Performance

19.2.2 Deterministic Assessment of Severe Accident Prevention

19.2.2.1.2 Mid-Loop Operation

During refueling or maintenance activities, the RCS is sometimes reduced to a “mid-loop” level. The applicant summarized the specific AP1000 design features that address mid-loop operations in DCD Tier 2, Section 5.4.7.2.1, “Design Features Addressing Shutdown and Mid-Loop Operations.” In addition, DCD Tier 2, Table 16.3-2, “Investment Protection Short-Term Availability Controls,” ensures that the RNS is available during mid-loop operation.

Section 19.3, “Shutdown Evaluation,” of this report documents the staff’s evaluation of shutdown risk. Because RTNSS has not been amended, it is not discussed in this supplement. DCD Tier 2, Table 16.3-2, documents the availability controls provided for the RNS during normal and reduced inventory. The staff concludes that the AP1000 design conforms to the mid-loop operation guidance specified in SECY-93-087 and is, therefore, acceptable.

19.2.3.3.1.3.2 Reactor Pressure Vessel Thermal Insulation System

Section 5.3.5 describes the design of the RPV thermal insulation system. Section 19.1.2.2.4 discusses considerations related to risk assessment. This section addresses the staff’s evaluation of conformance of the final design of the RVIS to the ULPU Configuration V testing, its integrity under the hydrodynamic loads associated with ERVC, absence of susceptibility to clogging, and absence of coatings that could interfere with wetting phenomena that contribute to effective heat removal.

In RAI-TR24-SPLA-01, the staff noted an apparent stepwise change in the cross-section of the annulus formed between the RVIS and the reactor vessel, formed by the neutron shield. This RAI also requested other information with respect to dimensions that could be important to the adequacy of the flow path. In a letter dated August 21, 2007, the applicant provided additional details about the modeling of the flow path and flow areas. This allowed the staff to confirm that the design was consistent with the ULPU Configuration V testing. The staff considers RAI-TR24-SPLA-01 resolved.

The applicant changed the design of the RVIS inlet closure devices from floating balls to hinged, buoyant doors. In RAI-TR24-SPLA-02, Part 1, the staff requested additional information to confirm that this was consistent with the ULPU Configuration V testing. In RAI-TR24-SPLA-07, the staff requested additional details on the configuration of the doors and the forces acting on them. In the August 21, 2007 letter, the applicant provided additional details about these active components of the system and the effect on flow areas and flow resistance. Because the new design retains the characteristic of actuation by buoyant forces, the cross-sectional area is maintained, and flow resistance is reduced, the staff considers this change to be an acceptable way to conform to the ULPU Configuration V testing.

Similarly, the applicant changed the design of steam vent ducts that provide a flow path for the steam/water within the reactor vessel insulation annular space to flow back to the containment flood-up region. In RAI-TR24-SPLA-02, Part 2, the staff requested that the applicant assess the impact of these design modifications. In the August 21, 2007 letter, the applicant explained that the previous design had multiple miter bends instead of a sudden contraction in the area of the flow path. The modification reduces flow resistance and allows higher mass flow. Because the results of the testing conservatively bound the expected performance of the RVIS, the staff considers this change to be an acceptable way to conform to the ULPU Configuration V testing. The staff considers RAI-TR24-SPLA-02 and RAI-TR24-SPLA-07 resolved.

The staff noted that the RVIS doors and the reactor coolant drain tank room ventilation damper, though described by the applicant as “passive,” are considered by the staff to be active components. In RAI-TR24-SPLA-03, the staff requested information on periodic verification of the performance of moving parts, and in RAI-TR24-SPLA-08, a discussion of as low as is reasonably achievable (ALARA) considerations for testing and maintenance was requested. In the August 21, 2007 letter, the applicant confirmed that RVIS is in the D-RAP. Proper fit and freedom of motion of the doors is confirmed during hot functional testing. Visual inspection and testing for freedom of rotation is to be performed during refueling outages at ten-year intervals, coordinated with other inspections in the same area. Individual doors and frames are designed for removal as a unit, so replacement, if required, would take little time. Because the applicant has applied a level of control and testing consistent with Commission policy on RTNSS and ALARA, the staff considers this to be acceptable and both RAI-TR24-SPLA-03 and RAI-TR24-SPLA-08 are resolved.

Because the design of the RVIS is consistent with the ULPU Configuration V testing, the staff’s evaluation of hydrodynamic loads and previous findings with respect to the potential for clogging of the flow path are unchanged.

In RAI-TR24-SPLA-06, the staff requested resolution of earlier test results (ULPU Configuration III and BETA tests) dealing with the wetability of the reactor pressure vessel surface if it were coated. The applicant revised the DCD to reflect a commitment to ensure that the reactor vessel exterior is bare metal.

On the basis of the additional description, the staff was able to confirm that the new design is consistent with the ULPU Configuration V ntesting and is, therefore, acceptable. The staff concludes that COL Action Item 19.2.3.3.1.3.2-1 is closed for COL applicants referencing the AP1000 DCD. The staff considers RAI-TR24-SPLA-06 resolved.

19.2.3.3.7 Equipment Survivability

Electrical and mechanical equipment must survive to prevent and mitigate the consequences of severe accidents. The applicant addressed equipment survivability in Appendix 19D, “Equipment Survivability Assessment,” to DCD Tier 2, which includes general requirements and equipment classification. Appendix D to the AP1000 PRA supporting document presents the analysis performed to determine the severe accident environmental conditions.

In TR-68, APP-GW-GLR-069, “Equipment Survivability Assessment,” May 2007, the applicant submitted revised analysis in support of the DC amendment. In reviewing TR-68, the staff noted that the severe accident environmental conditions were revised.

In APP-GW-VP-025, “AP1000 Equipment Survivability Assessment,” an attachment to TR-68, the applicant stated that it had revised the fraction of the core inventory released to the containment atmosphere from the original PRA Appendix D and that the values are consistent with the accident source term information presented in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants—Final Report.”

However, the staff identified certain analytical assumptions documented in APP-GW-VP-025 that did not appear to be consistent with NUREG-1465 and asked the applicant to clarify and confirm the basis for the results documented in TR-68. The NRC staff identified this as the first part of Open Item OI-SRP19.0-SPLA-14.

In a letter dated May 12, 2009, the applicant submitted additional information to resolve the perceived lack of consistency, and supplemented this with a letter dated September 8, 2009. The staff reviewed supporting calculations during an audit at the applicant's offices on October 13-15, 2009, documented in an audit report dated November 22, 2009. The staff is satisfied that the post-accident conditions were determined in a manner consistent with the applicable regulatory guidance, and the first of three parts of Open Item OI-SRP19.0-SPLA-14 is considered to be closed.

During the development of the severe accident management guidance (SAMG) for AP1000, additional requirements were defined for accident management in Time Frame 2 (in-vessel severe accident phase) and Time Frame 3 (ex-vessel severe accident phase). For example, previously unidentified methods of injecting water into the containment were added (e.g., providing makeup to overflow the IRWST by the RNS system). The use of containment spray was identified as a severe accident strategy (e.g., injecting water into containment and containment heat removal). Another method of depressurizing the RCS in Time Frame 2 was identified (i.e., reactor vessel head venting).

Finalizing certain system designs resulted in the need to update lists of associated equipment and instrumentation. For example, the applicant eliminated low-pressure steam generator feed systems (i.e., service water and condensate water) from consideration.

Lastly, the applicant changed the equipment and instrumentation identification to conform to updated naming conventions for AP1000. The new list of equipment and instrumentation reflects the amended AP1000 design.

The applicant has not completed the identification of equipment and instrumentation for prevention of core damage (e.g., Time Frame 0 and Time Frame 1) because the EOPs are still in development. Upon finalization of the EOPs, the applicant can identify and assess the survivability of the equipment and instrumentation used in those procedures. This should be identified as a licensee COL information item. The NRC staff identified this as the second part of Open Item OI-SRP19.0-SPLA-14.

In a letter dated May 12, 2009, the applicant submitted additional information that clarified the method by which this licensee COL information item is addressed. The staff finds that no additional holder item is required and, therefore, the second of three parts of Open Item OI-SRP19.0-SPLA-14 is considered to be closed. In general, the applicant claims that the AP1000 provides reasonable assurance that equipment, both electrical and mechanical, designed for mitigating the consequences of severe accidents will perform its functions as intended.

The staff questioned the completeness of the list of SSCs required for containment isolation. First, if a hydrogen monitor outside containment were required, additional penetrations might need to be included. In addition, it was not clear that the equipment survivability assessment needs to include mechanical penetrations and hatches (e.g., gasket materials) to ensure containment integrity. The NRC staff identified this as the third part of Open Item OI-SRP19.0-SPLA-14.

In a letter dated May 12, 2009, the applicant submitted additional information, and supplemented this with a letter dated September 8, 2009. The staff is satisfied that a hydrogen monitor outside containment was not included in the certified design and finds that there is no regulatory basis for requiring it. The applicant provided appropriate controls to ensure that gasket materials of mechanical penetrations and hatches will be capable of surviving post-accident conditions. In addition, the applicant has proposed an acceptable clarification of the DCD. The third of three parts of Open Item OI-SRP19.0-SPLA-14 is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19.2.5 Accident Management

Accident management encompasses those actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent core damage; (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel; (3) maintain containment integrity as long as possible; and (4) minimize offsite releases. Severe accident management is the process of extending the EOPs well beyond the plant design basis into severe fuel damage regimes and making full use of existing plant equipment, operator skills, and creativity to terminate severe accidents and limit offsite releases.

The NRC has taken an active role in ensuring that utilities adopt acceptable accident management practices. In January 1989, the staff issued SECY-89-012, “Staff Plans for Accident Management Regulatory and Research Programs,” which discusses essential elements of a utility accident management plan and offers an approach for accident management implementation. Subsequently, the NRC worked with the industry to define the scope and attributes of a utility accident management plan and to develop guidelines for plant-specific implementation. Section 5 of NEI 91-04, Revision 1, “Severe Accident Closure Guidelines,” which lays out the elements of the industry’s severe accident management closure actions that have been accepted by the staff, resulted from these efforts. This program involves the development of: (1) a structured method by which utilities may systematically evaluate and enhance their abilities to deal with potential severe accidents; (2) vendor-specific accident management guidelines for use by individual utilities in establishing plant-specific accident management procedures and guidance; and (3) guidance and material to support utility activities related to training in severe accidents. Using the guidance developed through this program, each operating plant has implemented a plant-specific accident management plan as part of an industry initiative.

Based on its reviews of these efforts, severe accident evaluations in individual plant examinations, and industry PRAs, the staff concluded that improvements to utility accident management capabilities could further reduce the risk associated with severe accidents. Although new reactor designs are to have enhanced capabilities for the prevention and mitigation of severe accidents, accident management remains an important element of defense in depth for these designs. However, the staff expects the increased attention on accident prevention and mitigation in these designs to alter the scope and focus of accident management relative to that for operating reactors. For example, the staff expects increased attention on accident prevention and the development of error-tolerant designs to decrease the need for operator intervention, while increasing the time available for such action if necessary. This permits a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

For both operating and advanced reactors, the overall responsibility for accident management, including development, implementation, and maintenance of the accident management plan, lies with the nuclear utility, because the utility bears ultimate responsibility for the safety of the plant and for establishing and maintaining an emergency response organization capable of effectively responding to potential accident situations. However, the vendors have played key roles in providing essential SAMG and strategies for implementation. This guidance has served as the basis for severe accident management procedures and for training utility personnel in carrying out the procedures. Computational aids for technical support have been developed, information needed to respond to a spectrum of severe accidents has been provided, decision making responsibilities have been delineated, and utility self-evaluation methodologies have been developed.

A COL applicant referencing the AP1000 design must develop and implement SAMG using the suggested framework provided in WCAP-13914, “Framework for AP600 Severe Accident Management Guidance,” Revision 3. This WCAP outlines a plan based on the Westinghouse Owners Group (WOG) SAMG for currently operating plants. Its scope is to address significant core damage accidents that might be possible in the AP600 and to provide the framework for developing guidance on how to cope with these accidents after the emergency response guidelines are no longer applicable. TR-66 extends this framework to the AP1000 design.

In NUREG-1793, the staff states that it expects the COL applicant to follow the recommendations provided in WCAP-13914 in developing its plant-specific accident management guidance (COL Action Item 19.2.5-1). The applicant has taken steps to facilitate this process by producing TR-66, the AP1000 framework document, and the AP1000 SAMG.

The applicant prepared and submitted TR-66 to close COL Information Item 19.59.10-4 with respect to development of the SAMG. The information item states the following:

> The combined license applicant referencing the AP1000 certified design will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914, “Framework for AP600 Severe Accident Management Guidance.”

APP-GW-GL-027, “Framework for AP1000 Severe Accident Management Guidance Development,” documents the framework for the AP1000. Based on this framework, the applicant has also developed SAMG for the AP1000 (APP-GW-GJR-400, “AP1000 Severe Accident Management Guidelines,” Revision A of January 2007), which COL licensees will implement at each site using the AP1000 design. TR-66 is a road map for COL licensees using these guidelines.

In RAI-SRP19.0-SPLA-15, the staff requested clarification of the schedule for development and implementation of SAMG. In a letter dated August 21, 2008, the applicant provided the requested information.

The starting point for the technical basis for the AP1000 is Electric Power Research Institute (EPRI) TR-101869, “Severe Accident Management Technical Basis Document” (Volumes 1 and 2). This document details the severe accident phenomenological understanding as it was in the early 1990s, when it formed the basis for the WOG guidelines. For the most part, this understanding is still current, although a number of important technical issues have been resolved and a few new ones identified since then. For example, direct containment heating in large-volume PWRs is no longer considered to be a major threat, but induced SGTRs in high-pressure scenarios have become a major concern. The AP1000 SAMG consists of three volumes:

1. An executive volume describes the methodology and criteria for the development of the AP1000 SAMG. It includes all of the material in the framework document, an overview of the AP1000 SAMG, a writer's guide for writing the SAMG and background documents, and a number of other important items related to the decision making process, interfaces between the AP1000 EOPs and the SAMG, and interfaces between the SAMG and the site emergency plan.

2. A guideline volume includes the SAMG guidelines to be used by the control room staff and the engineering support staff in the technical support center (TSC) in responding to a severe accident.

3. A background volume details the technical basis for the guidance found in the guidelines volume.

Section 5.1 of NEI 91-04 states that accident management consists of those actions taken during the course of an accident by the plant's emergency response organization (ERO), particularly the plant operations, technical support, and plant management staff, to achieve the following:

- prevent the accident from progressing to core damage
- terminate core damage progression once it begins
- maintain the capability of the containment as long as possible
- minimize onsite and offsite releases and their effects

The latter three actions constitute a subset of accident management referred to as severe accident management or, more specifically, severe accident mitigation.

NEI 91-04 also states that the goal of severe accident management is to enhance the capabilities of the ERO to mitigate severe accidents and prevent or minimize any offsite releases. The objective is to establish core cooling and to manage any current or immediate threats to the fission product barriers. Accomplishing this ERO should make full use of existing plant capabilities, including standard and nonstandard uses of plant systems and equipment.

The NRC staff agrees that NEI 91-04 properly defines the scope of severe accident management and believes that the framework for SAMG should be consistent with this scope.

The AP1000 SAMG consists of three parts: control room SAMG, TSC SAMG, and TSC challenge response guidance. The control room SAMG consists of two separate guidelines. The control room staff uses the first of these guidelines until the TSC is functional and its staff is ready to use the TSC SAMG. The staff uses the second guideline after the TSC is functional; this guideline provides the staff with a structured set of activities when the TSC is evaluating the plant conditions and potential responses.

The TSC staff will execute the TSC SAMG, using the diagnostic flow chart (DFC) and the severe challenge status tree to select the appropriate strategies to respond to variations in the key parameters. These strategies are described by the seven severe accident guidelines:

1. inject into containment
2. depressurize the RCS
3. inject into the steam generators
4. inject into the RCS
5. reduce fission product releases
6. control containment conditions
7. reduce containment hydrogen

The guidelines specify a method for a systematic, logical evaluation of the possible strategies and a process of deciding which actions to implement.

Another guideline, SAEG-1, monitors long-term activities after a particular strategy is implemented. Such activities depend on a number of factors, including the equipment put into service to implement the strategies, equipment already in service before implementing the SAMG that relates to the control of a DFC parameter, limitations on equipment usage identified in the guidelines that evaluate the possible strategies, equipment no longer in service if implementation of a strategy is discontinued, and changes in plant conditions following implementation of severe accident management strategies.

A final guideline, SAEG-2 for SAMG termination, comes into play when the plant has been put into a safe, stable state. At this time, selected parameters in the DFC are below their setpoint values and are stable or decreasing, and no new SAMG strategies will be required. However, generic SAMG exit guidance has been developed.

Four computational aids have been developed to assist the TSC staff in diagnosing and formulating appropriate strategies:

1. RCS injection to recover the core
2. injection rate for long-term heat removal
3. hydrogen flammability in containment
4. containment water level and volume

The NRC staff reviewed TR-66, the AP1000 framework document (APP-GW-GL-027), and the executive volume of the AP1000 SAMG. The staff confirmed that the AP1000 SAMG reflects current understanding of severe accident progression. The staff examined the remaining two volumes to ensure that they are an appropriate extension of the EPRI guidance, consistent with the SAMG developed by the WOG for operating reactors, and address all necessary high-level actions. The DCD appropriately references these documents.

Since concerns over hydrogen generation suggest maximizing the flow rate, while concerns about a degraded reactor vessel (overheating and wall thinning at or near the surface of the pool of relocated core material) suggest that flow should be controlled, in RAI-SRP19.0-SPLA-16, the staff asked the applicant to explain how it will provide guidance to resolve potentially conflicting considerations when introducing water to a dry reactor vessel after core relocation (full or partial) to the bottom head.

In a letter dated August 21, 2008, the applicant stated that it will update the AP1000 SAMG for “inject into the RCS” to address the recommended rate of injection into the RCS for situations in which the injection capability is recovered after significant core damage has occurred. The applicant will add a new item in the evaluation of the potential negative impacts of injecting into the RCS to note that high injection flow rates will minimize the potential for significant hydrogen generation while lower, controlled flow rates will minimize the potential for failure of the reactor vessel. The evaluation will explain that hydrogen generation caused by low flow rates will only be a concern if a significant amount of hydrogen is already in the containment indicating a failure of the hydrogen igniters or the recombiners, or both. On the other hand, the concern about reactor vessel integrity due to high injection flow rates will only exist when a prolonged period has elapsed since the onset of high core temperatures and the reactor vessel will be externally cooled by submergence in water.

Attachment B to the guideline and the associated background document will also provide a full discussion of the conditions under which each of the concerns is applicable. The staff considers RAI-SRP19.0-SPLA-16 resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

On the basis of this review, the staff concludes that SAMG for the AP1000 is consistent with NEI 91-04 and is a logical extension of the WOG SAMG. The discussions of the high-level actions, supported by the background documents provided, establish a sound technical basis for AP1000 COL applicants to develop their severe accident management procedures and training.

For this reason, the staff considers AP1000 COL Information Item 19.59.10-4 to be closed for COL applicants referencing the AP1000 DCD.

19.3 Shutdown Evaluation

19.3.7 Outage Planning and Control

In RAI-SRP19.0-SPLA-10, the staff asked the applicant to clarify whether the development of freeze seal guidelines is the responsibility of the COL applicant; is included in the procedure development described in TR-70, APP-GW-GLR-040, “Plant Operations, Surveillance, and Maintenance Procedures,” Revision 1, August 2007; or is controlled in some other way.

The applicant modified DCD Section 13.5 to include the following statement: “If freeze seals are to be used, plant-specific guidelines will be developed to reduce the potential for loss of RCS boundary and inventory when they are in use,” and confirmed that COL Information Item 13.5-1 includes the guidelines for use of freeze seals. It is among the guidelines identified as “Phase 3” procedure activities. The staff considers RAI-SRP19.0-SPLA-10 resolved.

NUREG-1793 described other COL action items related to shutdown procedures that the staff consolidated in COL Information Item 13.5-1. In RAI-SRP19.0-SPLA-11, the staff asked the applicant how it would identify these action items to the COL applicant:

- COL Action Item 19.1.8.1-4: The COL applicant will implement the maintenance guidelines as described in WCAP-14837, “AP600 Shutdown Evaluation Report.”

- COL Action Item 19.1.8.3-1: The COL applicant is responsible for developing procedures…to close containment hatches and penetrations following an accident during MODE 5 and MODE 6 before steam is released into the containment.

- COL Action Item 19.1.8.16-1: The COL applicant will have policies that maximize the availability of normal residual heat removal (valve V-023) and procedures to open this valve during cold shutdown and refueling operations when the RCS is open and the passive residual heat removal system cannot be used for core cooling.

- COL Action Item 19.1.8.16-2: The COL applicant will develop administrative controls to ensure that inadvertent opening of RNS valve V-024 is unlikely since inadvertent opening results in a draindown of the RCS inventory to the IRWST and requires gravity injection from the IRWST.

- COL Action Item 19.1.8.16-4: The COL applicant will maintain procedures to respond to low hot-leg level alarms.

- COL Action Item 19.3.7-1: The COL applicant will develop an outage planning and control program and will appropriately address the factors that improve low-power and shutdown operations consistent with DCD Tier 2, Chapter 19E, “Shutdown Evaluation,” and NUMARC 91-06, “Guidelines for Industry Actions To Assess Shutdown Management.”

- COL Action Item 19.1.8.7-1: The COL applicant will implement procedures and policies to have the nonsafety-related wide-range pressurizer level indication during cold shutdown.

Each of these action items has a corresponding entry in DCD Table 19.58-18. COL licensees referencing the AP1000 design must verify that the insights and assumptions documented in this table are satisfied. The staff finds this to be an acceptable method for ensuring that appropriate administrative controls will be applied. The staff considers RAI-SRP19.0-SPLA-11 resolved.

In TR-70, the applicant suggested that the procedures in Phase 3 need not be developed until after a COL is issued.

The staff agrees that these guidelines do not need to be completed at the time of application and finds that the program described in TR-70 is an acceptable method for providing assurance that appropriate guidance will be developed in a timely manner.

19.3.10 Flood Protection

NUREG-1793 states the following:

> The design provides fire detection and suppression capability. The design also provides flooding control features and sump level indication. The COL applicant is expected to take compensatory measures to maintain adequate detection and suppression capability during maintenance activities. This is part of COL Action Item 19.1.8.1-3.

The staff expects the COL licensees to take compensatory measures to maintain adequate detection capability during maintenance activities. The staff identified two COL information items that address fire detection and suppression but not flooding. In RAI-SRP19.0-SPLA-08, the staff asked the applicant to clarify how it will address concerns about flooding detection, barrier integrity, and control.

The applicant's August 21, 2008, response focused on aspects of the AP1000 design that obviate the need to compensate for a maintenance-related breach of flooding barriers. Specifically, the design does not include any watertight doors; flood barriers are permanent fixtures that are neither opened nor altered by normal activities, including maintenance. Moreover, the CDF contribution from internal flooding is extremely low.

The staff agrees with the applicant's assessment of internal flooding risk. As stated in NUREG-1793, the results of the AP1000 study for internal flooding show that the AP1000 design is adequate because internal floods during shutdown do not represent a significant risk contribution. The staff considers RAI-SRP19.0-SPLA-08 resolved.

19.5 Conclusion

The staff evaluated the information submitted by the applicant in accordance with NUREG-0800 Sections 19.0 and 19.1. In addition to its review of the documents identified above, the staff conducted a regulatory audit on August 9 and 10, 2010.

The applicant has updated the AP1000 PRA to include the most recent I&C design information. Additionally, to facilitate future updates, the applicant converted the PRA software package from the proprietary WesSAGE software package to the CAFTA software package. The applicant also documented the basis for its determination that other design changes did not affect the SSCs modeled in the PRA in a manner that affected the PRA.

In addition to reviewing the description of changes to the PRA, the staff reviewed the description of the new I&C design, specifically the PLS and PMS. The staff reviewed the methods and procedures for conversion of the PRA model from WesSAGE to CAFTA in the applicant's offices. The staff also reviewed the process by which the applicant incorporated the design changes in the PRA.

The staff noted that the applicant has a formal procedure for the review of design packages to ensure that it identifies and addresses any impact on the PRA. Design change packages are evaluated for the potential to alter the PRA model or to affect PRA assumptions or insights.

With the resolution of the open items described in Section 19.1.10 of this report, the staff concludes that the results and insights of the upgraded and updated design-specific PRA for the AP1000 demonstrate that the design meets the Commission's safety goals. These results and insights are an acceptable basis for the risk-informed review of the amended AP1000 DCD.

19F Aircraft Impact

This section describes the staff's evaluation of design features and functional capabilities of the AP1000 that are credited by the applicant to show that the facility can withstand the impact of a large commercial aircraft. These design features and functional capabilities were described in a letter dated March 19, 2010, that proposed changes to AP1000 DCD, Appendix 19F, "Malevolent Aircraft Impact." Upon reviewing the appendix, the staff found that the descriptions of design features and functional capabilities were incomplete. In RAI-SRP19F-AIA-01, the staff identified specific areas in the DCD that required augmentation. In response, the applicant proposed amendments to Appendix 19F and related sections of other DCD chapters.

The impact of a large commercial aircraft is a beyond-design-basis event. 10 CFR 50.150, "Aircraft impact assessment," requires applicants for new nuclear power reactors[7] to perform an assessment of the effects on the designed facility of the impact of a large, commercial aircraft. Applicants for DC are required to submit: (1) a description of the design features and functional capabilities identified as a result of the assessment in their DCD with (2) a description of how the identified design features and functional capabilities meet the acceptance criteria in 10 CFR 50.150(a)(1). Applicants subject to 10 CFR 50.150 must make the complete AIA available for NRC inspection in accordance with 10 CFR 50.70, "Inspections"; 10 CFR 50.71, "Maintenance of records, making of reports"; and Section 161.c of the Atomic Energy Act of 1954, as amended.

Regulatory Criteria

The staff used the following regulations and guidance to perform this review:

Regulations

10 CFR 50.150(a)(1) requires that applicants perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (i) The reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or spent fuel pool (SFP) integrity is maintained.

10 CFR 50.150(b) requires that the FSAR include a description of: (1) the design features and functional capabilities that the applicant has identified in accordance with 10 CFR 50.150(a)(1); and (2) how those design features and functional capabilities meet the assessment requirements of 10 CFR 50.150(a)(1).

Review Guidance

DG-1176 provides guidance for meeting the requirements in 10 CFR 50.150(a). It documents NRC endorsement of the methodologies described in the guidance prepared by NEI 07-13, Revision 7.

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[7] "Applicants for new nuclear power reactors" is defined in the Statement of Considerations for the Aircraft Impact Rule [74 FR 28112, June 12, 2009].

Supplementary information for the aircraft impact assessment rule [74 FR 28112], which indicates, among other things, that for the NRC to conclude that the rule has been met, it must find that the applicant has performed an AIA reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

The following are NRC staff interim review guidelines:

a. Reasonably Formulated Assessment

   The NRC considers an aircraft impact assessment performed by qualified personnel using a method that conforms to the guidance in NEI 07-13, Revision 7, to be a method which is reasonably formulated. The NRC considers qualified personnel to be: (1) an applicant who is the designer of the facility for which the AIA applies; and (2) an applicant's primary contractor for the aircraft impact assessment who has designed a nuclear power reactor facility either already licensed or certified by the NRC or currently under review by the NRC.

b. Reactor Core and Spent Fuel Pool Cooling Design Features

   The "reactor core cooling" criterion or "spent fuel pool cooling" criterion in 10 CFR 50.150(a)(1) is satisfied if design features have been included in the design of the plant to specifically perform that cooling function with reduced use of operator action.

c. Intact Containment

   The "intact containment" criterion in 10 CFR 50.150(a)(1) is satisfied if the containment: (1) will not be perforated by the impact of a large, commercial aircraft; and (2) will maintain ultimate pressure capability, given a core damage event until effective mitigation strategies can be implemented. Effective mitigation strategies are those that provide, for an indefinite period of time, sufficient cooling to the damaged core or containment to limit temperature and pressure challenges below the ultimate pressure capability of the containment as defined in Section 19 of the DCD, Revision 18.

d. Spent Fuel Pool Integrity

   The "spent fuel pool integrity" criterion in 10 CFR 50.150(a)(1) is satisfied if the impact of a large commercial aircraft on the SFP wall or support structures would not result in leakage through the SFP liner below the required minimum water level of the pool.

e. Reduced Operator Action

   The NRC considers use of operator action to be reduced when: (1) all necessary actions to control the nuclear facility can be performed in the control room or at an alternate station containing equipment specifically designed for control purposes; and (2) a reduced amount of active operator intervention, if any, is required to meet the acceptance criteria in 10 CFR 50.150(a)(1). Reduction in the use of operator action is measured relative to the actions required to address aircraft impact without the AIA rule in place (e.g., similar actions in operational programs in place at current operating reactor sites).

19F.1 Summary of Technical Information

Appendix 19F states that the applicant performed an AIA in accordance with the requirements in 10 CFR 50.150(a)(1) using the methodology described in NEI 07-13. Based on the results of the assessment, the applicant has identified a set of key design features to show that the acceptance criteria in 10 CFR 50.150(a)(1) are satisfied. These key design features are reported in Appendix 19F, along with references to other sections of the DCD that provide additional details. Appendix 19F also includes descriptions of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met.

19F.1.1 Description of Key Design Features

The key design features that are credited, their function(s), and references to sections including detailed descriptions of the features are summarized below:

Key Design Features

No.	Feature	DCD Reference	Function
1	structural design of the shield building	Chapter 3	protection of safety systems located inside containment
2	structural design of the auxiliary building	Chapter 3	protection of the SFP liner integrity
3	structural design of the wall along the south end of the turbine building at column line 11.2	Section 3.7.2.8.3	protection of the auxiliary building from the impact of a large, commercial aircraft
4	structural design of the wall along the east end of the annex building	Figure 3.7.2-19	protection of the auxiliary building from the impact of a large commercial aircraft
5	structural design and location of the SFP	Figure 3.7.2-12 Section 9.1.2.2	protection of the SFP from the effects of an impact of a large commercial aircraft
6	passive safety injection portions of the PXS and the ADS valves and spargers of the RCS inside containment	Section 5.4 Section 6.3	core cooling

Key Design Features

No.	Feature	DCD Reference	Function
7	physically separated locations of the main control room (MCR), remote shutdown station (RSS), and secondary DAS panel	Chapter 7	manual actuation of passive safety injection and recirculation for long-term core cooling can be initiated if required
8	supporting equipment required for operation of the squib valves from the MCR, RSS, or DAS panel including Class 1E batteries, control and instrumentation cabinets, cabling, and transfer switches	Section 7.7.1.11	manual actuation of passive safety injection
9	ADS Stage 4 squib valves IRWST injection line squib valves and recirculation line squib valves	Section 6.3	passive safety injection and recirculation for long-term core cooling
10	steel containment vessel	Section 19.40	containment integrity maintained with only air cooling for 24 hours
11	normal residual RNS containment isolation valves	Figure 5.4-7	isolation of the RNS outside of the containment
12	reactor trip equipment including sensors, manual inputs, protection and safety monitoring system cabinets, and reactor trip switchgear	Section 7.2.1	reactor shutdown
13	the design and locations of 3-hour fire barriers within the auxiliary building	Section 9.5.1	protection of equipment needed for manual actuation of systems and equipment potentially required for core cooling

Key Design Features

No.	Feature	DCD Reference	Function
14	specific barriers in the auxiliary building rated to withstand a differential pressure of 34.5 kPa (5 pound(s) per square inch differential (psid))	Section 9.5.1.2.1.1	limitation of the effects of fire damage created by the impact of a large, commercial aircraft

19F.1.2 Description of How Regulatory Acceptance Criteria Are Met

The acceptance criteria in 10 CFR 50.150(a)(1) are that: (1) the reactor core will remain cooled or the containment will remain intact; and (2) SFP cooling or SFP integrity is maintained. The applicant has met 10 CFR 50.150(a)(1) by including features in the AP1000 design that maintain: (1) an intact containment; and (2) SFP integrity following the impact of a large commercial aircraft.

The applicant credits the shield building as a structure that will remain intact following an impact by a large commercial aircraft. Therefore, containment will also be intact. The RPV, PXS, and equipment within the containment will not be damaged by the impact or by exposure to jet fuel. For a postulated impact on the auxiliary building, the applicant credited the design and locations of 3-hour rated fire barriers to limit damages such that manual actuation of core cooling equipment, if necessary, can be achieved.

The AP1000 design also satisfies the SFP integrity acceptance criterion because it is surrounded by barrier walls that protect the SFP liner from adverse effects of an impact by a large commercial aircraft.

19F.2 Evaluation

The staff has reviewed the description of key design features provided by the applicant and the description of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. The staff's evaluation is provided below.

19F.2.1 Reasonably Formulated Assessment

The applicant states that its AIA is based on the guidance of NEI 07-13, Revision 7. In a letter dated August 6, 2010, in response to RAI-SRP19F-AIA-07, the applicant confirmed that the AP1000 assessment does not take exceptions to portions of NEI 07-13 guidance that apply to the AP1000 design. In a revised RAI response dated September 15, 2010, the applicant further stated that an analytical evaluation and experimental verification has been performed for the first-of-a-kind steel-concrete modular design feature subjected to the aircraft impact loading in accordance with the recommendation set forth in Section 2.4.1(4) of NEI 07-13. Based on the applicant's use of NRC-endorsed guidance document, NEI 07-13, Revision 7, by qualified personnel, the staff finds that the applicant has performed a reasonably formulated assessment.

19F.2.2 Key Design Features for Core Cooling

The staff found the initial description of design features and functional capabilities for core cooling to be incomplete. In RAI-SRP19F-AIA-01, the staff identified specific areas in the DCD that needed to be augmented with additional information. The staff's review of the applicant's response to RAIs pertaining to core cooling is discussed below.

In RAI-SRP19F-AIA-01, the staff requested that the applicant identify and describe the specific design features relied upon to maintain core cooling following impact of a large commercial aircraft during power operation. In its response dated September 17, 2010, the applicant identified the following key design features: safety-related passive safety injection and long-term recirculation cooling systems, specific squib valves that need to actuate and equipment that supports actuation of the squib valves, including Class 1E batteries, control and instrumentation cabinets, cabling, and transfer switches. These design features are fully described in DCD Section 6.3 and parts of Section 7.

The staff reviewed the descriptions of these design features and found that the passive safety injection and long-term recirculation cooling system, have been designed specifically to maintain core cooling functions following design-basis events initiated during power operation. The staff considered the descriptions of the features, as well as the ability of these features to perform their design basis safety functions following impact of a large commercial aircraft, including conditions involving loss of coolant from the RCS, and finds that they are suitable for maintaining core cooling following an impact of a large commercial aircraft,. The staff also considered that since the AP1000 AIA credited the shield building to remain intact upon an aircraft impact, the components of the passive safety injection and long-term recirculation systems that are located within the containment structure would not be exposed to jet fuel damage. Furthermore, should this design feature fail to initiate automatically, it can be initiated or operated either from the MCR, the RSS, or the secondary DAS panel, and require no further operator intervention to maintain the core cooling function. This function can be achieved with the key design features identified in the table above as Numbers 1, 6, 8, and 9. On this basis, the staff finds the applicant's description of the key design features for core cooling to be adequate. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In RAI-SRP19F-AIA-02, the staff requested that the applicant identify and describe the specific design features that are relied on to maintain core cooling following impact of a large commercial aircraft while the plant is shut down and the reactor is being cooled by the normal RNS. In its response dated August 6, 2010, the applicant states that should this occur, one of two sets of RNS containment isolation valves located inside containment can be closed from the MCR to terminate leakage from the RCS. These valves and the spatial separation between their location and the location of the MCR are identified as key design features. Core cooling is provided by gravity injection from the IRWST, initially, and the containment recirculation system in the long term. These two systems, including their squib valves, are identified in the DCD as key design features.

AP1000 DCD Section 19.E.2.3.3.1 describes use of these features for core cooling following a loss of RNS during shutdown in MODE 5 with the RCS open. The staff's review of this section of the DCD is described in Chapter 19 of NUREG-1793. The applicant has identified appropriate design features that have been shown to be effective in providing core cooling when the RNS is not available. Based on the above, the staff finds the applicant's description of the key design features for core cooling to be adequate. In a subsequent revision to the

AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

To conform to the guidance in NEI 07-13, applicants must consider whether the impact of large commercial aircraft would cause damage that could prevent the reactor from shutting down in the unlikely event that operators did not manually trip the reactor prior to impact. In RAI-SRP19F-AIA-03, the staff requested that the applicant describe those design features that assure the reactor will be shut down following an aircraft impact, including any features that protect equipment needed for reactor shutdown. In its response dated August 6, 2010, the applicant stated that the equipment needed for reactor shutdown is described in DCD Section 7.2.1 and is considered a key design feature(s). They also indicated that the design of this equipment is such that the reactor trip function will fail in a safe manner (control rods drop into the reactor by the force of gravity) if any of this equipment should become damaged by an aircraft impact. The fail-safe design of this equipment is described in DCD Section 7. Based on the above the staff finds that the applicant has adequately described how the reactor will be shut down should equipment normally relied upon for reactor shutdown be damaged by aircraft impact. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In its initial submittal, the applicant identified the locations of the MCR, RSS and the secondary DAS panel as key design features. In RAI-SRP19F-AIA-04, the staff requested that the applicant describe the roles these features play in satisfying the acceptance criteria in the rule. The staff also requested that the applicant clarify which SSCs in these locations are being credited for actuation of the core cooling equipment. In its response dated August 6, 2010, the applicant stated that manual initiation of core cooling requires operator action from the MCR, RSS, or secondary DAS panel and that each location contains equipment capable of initiating core cooling design features. The applicant also stated that the AIA shows that no single strike from a large commercial aircraft can simultaneously damage the equipment at all three locations. Thus, the degree of separation between these specific facilities is a key feature of the plant design that enables the core cooling acceptance criterion in the rule to be met. The applicant included similar statements in DCD Appendix 19F.4.2. The applicant also added a description of the equipment in these locations needed to actuate core cooling design features. This equipment includes the Class 1E batteries, the supporting PMS control and instrumentation cabinets and cabling for the equipment identified in Appendix 19F.4.3, the transfer switch to isolate the MCR and transfer controls to the remote shutdown room, and the DAS cabling for the squib valve control cabinet.

The staff reviewed detailed descriptions of equipment needed for safe shutdown of the facility in DCD Section 7 and compared them with the set of SSCs cited by the applicant as needed to actuate core cooling design feature. The staff found that the applicant has adequately described key design features needed for actuating core cooling. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

19F.2.3 Key Design Features that Protect Core Cooling Design Feature

Structures

In Appendix 19F, the applicant states that the robust shield building design, as described in DCD Chapter 3, is a key design feature that would provide protection for the core cooling key design feature. It also states that the AP1000 assessment concluded that a strike upon the shield building by a large commercial aircraft would not result in perforation of the shield building and the containment vessel. Because the passive safety injection and long-term recirculation cooling systems that are credited as key design features for core cooling are located inside the containment vessel, it is expected that they will not be damaged, either from an aircraft impact or from exposure to jet fuel.

Fire Protection

The key design features that protect the core cooling key design feature also include the 3-hour rated barriers around and within the auxiliary building as described in DCD Sections 9.5.1 and 9A. The design and locations of fire barriers within the auxiliary building are credited to confine the spread of fire damage resulting from a large commercial aircraft impact. The applicant also credited 5 specific 34.5 kPa (5 psid) fire barriers, as described in DCD Section 9.5.1.2.1.1, to further limit fire spread. The AP1000 assessment determined that no aircraft impact scenario would cause perforation and subsequent fire propagation into the containment where the core cooling equipment is located. Neither would any scenario simultaneously destroy all three redundant locations where support equipment for manual actuation of the core cooling function is located. These key design features are identified in the table above as Numbers 1, 7, 13, and 14. On this basis, the staff finds the applicant's description of the key design features for protecting safety systems required to maintain core cooling to be adequate, as these systems are physically separated and protected by robust structural barriers.

19F.2.4 Containment Intact

The applicant states that the shield building and the protected steel containment vessel as described in DCD Chapter 3 are key design features for maintaining containment intact. Appendix 19F also states that the AP1000 assessment concluded that an aircraft strike upon the shield building would not result in perforation of the shield building and, therefore, the steel containment vessel is not affected. Based on the AP1000 beyond design basis calculation, air-only cooling of containment is sufficient to allow containment integrity to be maintained for 24 hours. This capability is achievable by key design features 1 and 10 as described above. Based on the above, the staff finds the applicant's description of the key design features for maintaining containment intact to be adequate as the containment: (1) will not be perforated by the impact of a large, commercial aircraft; and (2) will maintain ultimate pressure capability.

19F.2.5 Integrity of the Spent Fuel Pool

The key design features credited to maintain the integrity of the SFP are its location and structural design, as described in DCD Section 9.1.2.2 and Figure 3.7.2-12. The applicant indicates that the location and design of the SFP structure are such that it can withstand the effects of an impact of a large, commercial aircraft. The staff finds the applicant's description of the key design features for ensuring SFP integrity to be adequate.

19F.3 Conclusion

The staff finds that the applicant has performed an AIA that is reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the acceptance criteria in 10 CFR 50.150(a)(1) are met. The staff finds that the applicant adequately describes the key design features and functional capabilities credited to meet 10 CFR 50.150, including descriptions of how the key design features show that the acceptance criteria in 10 CFR 50.150(a)(1) are met. Therefore, the staff finds that the requirements of 10 CFR 50.150(b) are met.

22. REGULATORY TREATMENT OF NON-SAFETY SYSTEMS

22.5.6 Post-72-Hour Actions and Equipment

The staff issued a request for additional information (RAI) on equipment that may be needed 72 hours after an accident (RAI‑SRP19.0‑SPLA‑20). Westinghouse Electric Company, LLC, (the applicant) responded in a letter dated July 15, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091980042), clarifying the description of ancillary equipment for the AP1000.

The AP1000 design relies on the following safety functions 72 hours after an accident:

- core cooling, inventory, and reactivity control
- containment cooling and ultimate heat sink
- main control room habitability
- post-accident monitoring
- spent fuel pool cooling

To support these functions, the AP1000 design includes nonsafety-related ancillary equipment. Connections are provided for generators and pumping equipment that can be brought to the site to back up the installed equipment. The ancillary equipment (alternatively, the transportable equipment) is capable of supporting extended operation of the passive safety systems.

- Electrical power is required to supply the following loads:

  – post-accident monitoring instrumentation

  – spent fuel pool monitoring instrumentation

  – ventilation for the main control room, instrumentation and control room, and direct current equipment room

  – power to replenish the passive containment cooling water storage tank (PCCWST) using motor-driven pumps

- After the energy stored in the 1E batteries is depleted, ancillary diesel generators in the annex building provide alternating current (ac) power. A fuel tank stores sufficient fuel for 4 days of operation (both trains). Power from each ac generator is fed to a distribution panel, from which it supplies the associated 1E battery recharger and passive containment cooling system (PCS) recirculation pump, as well as local services (heating for the ancillary diesel fuel tank and lighting for the ancillary diesel space). All of this equipment is in the seismically qualified portion of the annex building, which is also designed to withstand high winds and associated missiles.

- The seismically qualified portion of the annex building is accessible through the auxiliary building, which is a safety-related, seismic Category I structure. Seven days are available for plant operators to restore at least one path to the outside, through which the fuel tank can be refilled or a transportable generator can be connected to the distribution panel described above.

- Makeup water is required for the PCCWST, which provides water for the following:

  - containment cooling
  - firefighting
  - spent fuel pool cooling by maintaining the inventory of water in the pool

- The initial inventory of the PCCWST is adequate for 72 hours. An additional 4-day supply of water is stored in the passive containment cooling ancillary water storage tank, which is a seismic Category II structure, and designed to withstand high winds and associated missiles. One of two recirculation pumps in the PCS pumps this water to the PCCWST. If normal power is not available, an associated ancillary diesel generator can power each of the PCS recirculation pumps.

- A nonsafety-related connection is provided for external makeup to the PCCWST. It is compatible with available firefighting equipment and accessible from the yard outside the auxiliary building ("plant west" of the auxiliary building). Through this connection, any additional water required can be injected at the safety-related return line from the PCS recirculating pump and heater to the PCCWST.

The staff concludes that the design features described above provide adequate assurance that required safety functions can be maintained in the long term (beyond 72 hours post-accident) and after seismic events. This is consistent with the Commission's staff requirements memoranda on SECY-94-084 and SECY-95-132, both titled "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994, and May 22, 1995, respectively, and NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Therefore, the design described above is acceptable.

22.5.9 Short-Term Availability Controls

AP1000 design control document (DCD) Tier 2, Table 16.3-2, "Investment Protection Short-Term Availability Controls," identifies short-term availability controls for nonsafety-related structures, systems, and components (SSCs) that are subject to regulatory treatment.

There are no limiting conditions for operation if the completion times for required actions are not met (i.e., there is no requirement to bring the plant to a safe-shutdown condition when operability requirements are not fulfilled). The staff finds this acceptable since these nonsafety-related systems do not meet any of the four criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c)(2)(ii), "Technical specifications," that would require a limiting condition for operation:

(1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary

(2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier

(3) an SSC that is part of the primary success path and that functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier

(4) an SSC that operating experience or probabilistic risk assessment has shown to be significant to public health and safety

In addition, inspections, tests, analyses, and acceptance criteria, (ITAAC) as described in DCD Tier 1, Section 3.7, "Design Reliability Assurance Program," address these SSCs. They are described in DCD Tier 2, Section 17.4, "Design Reliability Assurance Program," and identified in Table 17.4-1, "Risk-Significant SSCs Within the Scope of D-RAP"; therefore, the staff finds the administrative controls for regulatory treatment of nonsafety systems in DCD Tier 2, Table 16.3-2, to be acceptable.

In DCD Tier 2, Section 16.3.2, "Combined License Information," the applicant stated that combined license (COL) applicants referencing the AP1000 will develop a procedure to control the operability of investment protection SSCs in accordance with DCD Tier 2, Table 16.3-2. In DCD Tier 2, Section 13.5, "Plant Procedures," the applicant described the commitment to address operational and maintenance programmatic issues. The staff finds that COL Information Item 13.5-1 provides an acceptable method for ensuring that licensees will develop a procedure to control the operability of nonsafety-related SSCs subject to regulatory treatment.

23. DESIGN CHANGES PROPOSED IN ACCORDANCE WITH ISG-11

Westinghouse has submitted information in support of its design certification (DC) amendment application that Westinghouse considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5). Westinghouse has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears within "square brackets" as follows:

[   ]

The complete text of this chapter, including proprietary information, can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

23. Introduction

This chapter addresses new design changes proposed by Westinghouse (the applicant) that were included in Revision 18 of the AP1000 design control document (DCD). The design changes that are evaluated in this chapter do not constitute all of the changes that the applicant proposed to include in the DCD, Revision 18. Rather, the design changes evaluated in this chapter are in addition to those that the applicant has submitted to the NRC as responses to requests for additional information (RAIs) or safety evaluation report (SER) open items.

The applicant proposes that the DCD be changed by adding new, more detailed design information that expands upon the design information already included in the DCD. This information would be used by every combined license (COL) application that references the certified AP1000 design. The regulations of 10 CFR 52.63(a)(1), "Finality of standard design certifications," require that the Commission may not modify a DC, whether on its own motion or in response to a petition from any person, unless the Commission determines that the change meets one of seven criteria. The staff finds that each of the changes evaluated in this chapter meets one or more of the seven criteria in 10 CFR 52.63(a)(1). Further, the staff finds that many of the design changes evaluated in this chapter contribute to the increased standardization of the certification information because the design amendment would be applied to all COL applicants that reference the DC rule. Therefore, these changes enhance standardization, and meet the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

Interim Staff Guidance, DC/COL-ISG-011, "Interim Staff Guidance Finalizing Licensing-basis Information," describes the staff's position regarding the control of licensing-basis information during and following the initial review of applications for DCs. In part, DC/COL-ISG-11 describes the categories of design changes that applicants should not defer until after the issuance of the DC rule. Categories of those changes that should not be deferred include:

- the correction of significant errors in an application;
- changes needed to ensure compliance with NRC regulations;
- changes needed to support other licensing-basis documents (e.g., conforming changes to information in the final safety analysis report (FSAR) supporting technical specifications (TS));
- significant technical corrections associated with the design or program described in the licensing document (i.e., if not changed, would preclude operation within the bounds of the licensing basis, as opposed to proposed alternatives to the described design or program); and
- changes needed to address a significant vulnerability identified by probabilistic risk assessments (PRAs) or other studies (e.g., a change in a PRA insight).

23.A Changes to Component Cooling Water System

23.A.1 Description of Proposed Changes

In letters dated April 26, 2010, and July 29, 2010, the applicant proposed changes to the design of the component cooling water system. The normal residual heat removal system (RNS), component cooling water system (CCS) relief valves increase in size from 2.54 centimeters (cm) x 2.54 cm (1 inch (in) x 1 in) to 7.62 cm x 10.16 cm (3 in x 4 in) to meet required flow capacity requirements. In addition, the CCS surge tank vent-line increases in size from 5.08 cm x 7.62 cm (2 in to 3 in) nominal pipe size (NPS). Tier 2 text in the DCD is modified to include Sections 3.4.1.2.2.2, “Auxiliary Building Flooding Events”; 9.2.2.3.4, “Component Cooling Water System Valves”; 9.2.2.4.5.2, “Leakage into the Component Cooling Water System from a High Pressure Source”; 19E.2.5, “Component Cooling and Service Water Systems”; and Figure 9.2.2-2, “Component Cooling Water System Piping and Instrumentation Diagram.”

23.A.2 Regulatory Basis

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.” While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the reactor coolant system (RCS) is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to regulatory treatment of nonsafety systems (RTNSS) in accordance with the Commission’s policy for passive reactor plant designs. The staff’s evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related structures, systems, and components (SSCs) or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of inspections, tests, analyses and acceptance criteria (ITAAC), test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for

Nuclear Power Plants," Section 9.2.2, "Reactor Auxiliary Cooling Water System," Revision 4, March 2007, as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified in NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.A.3 Evaluation

During the staff's evaluation of the proposed CCS design changes, it was determined that additional information was required from the applicant. The staff generated RAI-DCP-CN-06-SBP-01 addressing three issues.

Information previously supplied in a letter dated January 20, 2010, for Section 15.6.2, "Failure of Small Lines Carrying Primary Coolant Outside Containment," was omitted from the April 26, 2010 submittal. This information appeared to be relevant but was removed without an explanation. Additional information is needed to justify that a Chapter 15 change is no longer needed.

The basis for the increased CCS surge tank vent line from 5.1 cm x 7.6 cm (2 in to 3 in) (overflow protection due to normal residual heat removal system leakage into CCS) was not described in DCD Section 9.2.2. The applicant should consider adding this information to the DCD.

The flow rate (as measured in liters per minute (Lpm) (gallons per minute (gpm)) of the CCS/RNS relief valve(s) through the waste water system (WWS) was not described as related to the capacity of the auxiliary building sump pumps (concerns for potential building flooding if the relief valve flow rate exceeds the sump pump flow rate). The applicant should consider adding this information to the DCD.

The applicant's response to RAI-DCP-CN-06-SBP-01 provided the following:

1. The information was deleted from this section in the final proposed change because it describes a non-limiting case (i.e., it is bounded by the sample line break). Discussion of the RNS heat exchanger tube leak has been incorporated into Appendix 19E, Section 19E.2.5.

2. The CCS surge tank line vent/overflow line was increased in size from 5.1 cm x 7.6 cm (2 in to 3 in) to eliminate the potential for over-pressurizing the surge tank (designed as an atmospheric tank) in the event of a large RNS heat exchanger tube leak that causes a significant increase in liquid volume in the CCS. The basis for the increase in surge tank vent line size is described in the proposed revision to DCD Section 9.2.2.3.3.

3. The maximum flow rate possible as a result of a double-ended break of one RNS heat exchanger tube is approximately 1968 Lpm (520 gpm). The large relief valve on the RNS heat exchanger cooling water line discharges to the radioactive waste drain system (WRS) auxiliary building equipment and floor drain sump at elevation 20.3 meters (m) (66 feet (ft)-6 in). This sump is pumped to the waste holdup tank by two air-driven sump pumps, each of which has a nominal capacity of 473 Lpm (125 gpm). In the event that the relief valve discharges continuously for an extended period of time, the WRS floor drain sump may overflow into the 20.3 m (66 ft-6 in) level of the auxiliary building. Section 3.4.1.2.2.2 in Tier 2 of the DCD describes auxiliary building flooding events. In the radiologically controlled area of the first level (elevation 20.3 (66 ft-6 in)) there are no

safe shutdown components and the maximum flood elevation has been determined to be 30.5 cm (12 in) or less, assuming that the flooding is identified and isolated within 30 minutes.

Based on the staff's review, the applicant's response for Item 1 was determined to be acceptable since the DCD Tier 2 text that was deleted for Section 15.6.2 (between the January and the April letters) was not a bounding case and did not need to be included as part of DCD Section 15.6.2. The information that was added to Section 19E.2.5 was evaluated by the staff and was found acceptable since the DCD markup added relevant information associated with the RNS heat exchanger and overpressure protection. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

For Item 2, the staff finds the applicant's response to be acceptable since the increase in relief valve size assures that the CCS surge tank does not over-pressurize due to an RNS heat exchanger tube leak. The existing 5.1 cm (2 in) NPS was too small to pass the approximate 1968 Lpm (520 gpm) leakage from the RNS tube rupture. The DCD markup was provided to add this RNS leak-rate value to DCD Tier 2, Section 9.2.2.3.3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

For Item 3, the staff finds the applicant's response to be acceptable since an overflow of the WRS floor drain sump does not affect safe shutdown components of a RNS heat exchanger tube break. The maximum flooding level on this floor (auxiliary building, elevation 20.3 (66 ft-6 in) was determined to be 30.5 cm (12 in) or less above the floor, assuming flooding is identified and isolated within 30 minutes.

23.A.4 Conclusion

The staff's review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable. RAI-DCP-CN-06-SBP-01 is resolved.

23.B Changes to Component Cooling Water System

23.B.1 Description of Proposed Change

In letters dated April 26, 2010, and July 29, 2010, the applicant proposed changes to the design of the CCS. Four piping header systems were added for the RNS/CCS relief valves. DCD Tier 2, Figure 9.2.2-2, was modified to include these piping header systems.

23.B.2 Regulatory Basis

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified by NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.B.3 Evaluation

During the staff's evaluation of the proposed CCS design changes, it was determined that additional clarification to the description of the design changes was required from the applicant; therefore, the staff generated RAI-DCP-CN-09-SBP-01.

The applicant's response to RAI-DCP-CN-09-SBP-01 is as follows:

> The "Eliminate the RNS …" statement should be replaced with "Larger pressure relief valves have been added to the CCW system." The reason for this design change is further clarified by:
>
> The original (1" x 1") thermal relief valves provided for the RNS and SFS heat exchangers were intended to discharge to the floor of the CCS valve room, where the discharge would be collected by a nearby floor drain. Now that much higher capacity relief valves are needed for the RNS heat exchanger cooling water lines to meet ASME VIII equipment overpressure protection requirements, the larger valves needed for the RNS heat exchanger (V302A/B) are piped to a dedicated drain collection funnel located in the valve room to minimize the potential release of large quantities of vapor and water spray to the room in the event that a high capacity discharge occurred. The smaller valves (V342A/B) are also provided with their own collection header that discharges near the existing room floor drain. These changes are shown in the revised sheet 3 of Tier 2 DCD Figure 9.2.2-2.

Based on the staff's review, the applicant's response was determined to be acceptable since the design includes a collection piping header and dedicated drain funnel to handle the potential release of large quantities of vapor and water spray to the room in the event of a high capacity relief valve discharge. This discharge connects to the main drain header in the auxiliary building. The previous design allowed relief valve discharge to the floor with water collection to the nearest floor drain. The DCD markup was provided to add this design improvement to DCD Tier 2, Figure 9.2.2-2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.B.4 Conclusion

The staff' concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable. RAI-DCP-CN-09-SBP-01 is resolved.

23.C Changes to Component Cooling Water System

23.C.1 Description of Proposed Changes

In a letter dated June 18, 2010, the applicant proposed changes to the design of the CCS. The proposed changes increase the size of the relief valves on four cooling water lines associated with the reactor coolant pumps (RCPs), and one chemical and volume control system (CVS). Specifically, the RCP cooling water line relief valves V253/A/B/C/D are changed from 7.6 cm x 10.2 cm (3 in x 4 in) to 10.2 cm x 15.2 cm (4 in x 6 in) with the associated branch lines L253A/B/C/D changed from 7.6 cm (3 in) to 10.2 cm (4 in). Also the CVS letdown cooling water line relief valve V222 is changed from 5.1 cm x 7.6 cm (2 in x 3 in) to 7.6 cm x 10.2 cm (3 in x 4 in) with the associated branch line size of L222 changed from 5.1 cm (2 in) to 7.6 cm (3 in). The basis for the change is to prevent over-pressurization of the CCS piping systems. DCD Tier 2, Figure 9.2.2-2 was modified to include these changes.

23.C.2 Regulatory Basis

The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes will not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified in NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

23.C.3 Evaluation

Based on the staff's review, these design changes were determined to be acceptable. The changes to the five relief valves and associated branch piping are necessary to prevent over-pressurization of the CCS due to a postulated tube leak in the RCP and CVS heat exchangers with the cooling water lines isolated. The larger relief valves, 10.2 cm x 15.2 cm (4 in x 6 in) for the RCPs and 7.6 cm x 10.2 cm (3 in x 4 in) for CVS, and associated piping have been adequately sized for the required relief flow, thus preventing possible damage to the CCS. The DCD markup was provided to address these changes to DCD Tier 2, Figure 9.2.2-2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.C.4 Conclusion

The staff's review concludes that the design changes described above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. These design changes were evaluated with respect to the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS and found acceptable.

23.D Changes to Ancillary Diesel Generator System

23.D.1 Description of Proposed Changes

In letters dated May 10, 2010, July 29, 2010, and August 5, 2010, The applicant proposed changes to the design of the ancillary diesel generator system. These design change apply to the nonsafety-related ancillary diesel generators and include the following changes:

- Increase the rating (from 35 kilowatts (kW) to 80 kW) and physical size of the ancillary diesel generators
- Increase the size of the diesel fuel oil storage tank and change principal construction code to the American Society of Mechanical Engineers (ASME) Code, Section VIII
- Revise certain circuit breaker sizes
- Revise corresponding physical drawings and electrical one line diagrams
- Revise design of the heating, ventilation, and air conditioning (HVAC) system for ancillary diesel generator and oil storage tank rooms

23.D.2 Regulatory Basis

Sections 8.3.1.3 and 9.5.4 of NUREG-1793 address the function and acceptability of the ancillary diesel generators for the AP1000. No regulatory basis documents are identified as applicable to the ancillary diesel generators since they are nonsafety-related, commercial grade equipment. Acceptability of these design changes were based on conformance with the existing AP1000 licensing basis.

23.D.3 Evaluation

The ancillary diesel generators are designed to provide the post-72 hour power requirements following an extended loss of onsite power sources. The ancillary diesel generators are not safety-related but provide electric power for Class 1E post-accident monitoring, main control room (MCR) lighting, MCR and Division B and C instrumentation and controls (I&C) room ventilation and for refilling the passive containment cooling system (PCS) water storage tank and the spent fuel pool (SFP) when no other sources of power are available. The ancillary diesel generators are not required to perform these functions for the first 72 hours following a loss of all other alternating current (ac) power sources. As stated in the proposed design changes submittal, the design of the anchorage for these skid-mounted package units is consistent with the safe-shutdown earthquake (SSE) design of equipment anchorages of seismic Category II equipment. The off-skid fuel oil piping and fuel oil storage tank are analyzed to show that they withstand an SSE.

During the staff's evaluation of the proposed design changes to the ancillary diesel generator design, it was determined that additional information was required from the applicant to complete the staff's evaluation. The staff generated RAI-DCP-CN-59-SBP-01 requesting clarification of the design changes and justification for the changes to the ventilation system for cooling the diesel. In addition, the description of the design changes did not appear to be consistent with the changes made to the DCD description for this system.

The heat removal arrangement for the operating diesels was changed to include ducting from the discharge of the diesel radiators through a backdraft damper to a hurricane-proof louver on an exterior wall of the diesel enclosure building. There is also a damper and recirculation bypass in the cooling exhaust duct that can be closed in cold weather to recirculate engine heat to the diesel generator room for room heating.

The inlet cooling air is admitted to the diesel generator room by opening a personnel access door on an exterior wall of the enclosure building when the diesel generator is operating. The staff questioned the inconsistency between an exhaust protected against hurricanes and an intake that is not protected during operation of the diesel generator.

The applicant response to RAI-DCP-CN-59-SBP-01 provides sufficient clarification of the design change to support the staff's evaluation and also provides proposed additional changes to the DCD to document the clarifications. With respect to the design for hurricane conditions, the applicant stated that a hurricane is only considered to be an initiating event and is postulated to occur when the ancillary diesel generators are not in operation. When the ancillary diesel generators are not operating, the exterior door used to provide intake of engine cooling air will be closed and this door is designed to withstand a hurricane, including windborne missiles. Since no more than a single initiating event hurricane is postulated for the design basis, hurricane protection is not needed when the ancillary diesel generators are operating and the door that provides inlet cooling air is open.

The change in design code for the ancillary diesel fuel oil storage tank from Underwriters Laboratories (UL) to ASME Code Section VIII provides a more rigorous and extensive set of design, fabrication and testing requirements for the tanks, and is, therefore, acceptable to the staff.

The proposed changes to the ancillary diesel generator do not change the ancillary diesel generators' physical or functional relationship to safety-related SSCs and, therefore, do not increase the potential to adversely affect these SSCs. In addition, the proposed DCD changes include a statement that the ancillary diesel generators and associated SSCs are designed to preclude spatial interaction with any other nonseismic SSC that could adversely interact to prevent the functioning of the post-72 hour SSCs following an SSE. The proposed DCD changes as revised in the response to RAI-DCP-CN-59-SBP-01 are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant proposed to change the size of the ancillary generator from 35 kW to 80 kW because the starting motor current of the PCS recirculation pump, which can draw up to 6.5 times the full load current, was not considered in the generator sizing calculations. In addition, as part of this modification, the applicant proposed to revise the input circuit breaker size for the regulating transformer from 20 amps to 125 amps, the load test breaker from 100 amps to 60 amps, the PCS pump motor from 20 amps to 50 amps, revise Figure 8.3.1-3 of DCD Tier 2, and update Class 1E 208/120 uninterruptible power supply (UPS) one line diagram (Figure 8.3.2-2) to change "Transportable" AC Generator to "Ancillary" AC Generator. In addition, the applicant proposed other associated physical arrangement drawing changes to accommodate the increased size of the ancillary generator.

The staff reviewed the information provided by the applicant and was concerned that the revised rating of the distribution panel and the size of the output breaker for load testing of the ancillary generator were not compatible to accommodate the higher output current from the upgraded ancillary generator. In addition, the staff found that the proposed change to revise the input circuit breaker size for the regulating transformer from 20 amps to 125 amps did not seem to be adequate. In RAI-SRP 8.3.1-EEB-2, the staff requested the applicant provide its basis for the following:

(1) Utilizing a 125 amp breaker to protect 45 kilovolt amps (kVA) regulating transformer.

(2) Adequacy of the distribution panel rating of 100 amps for the 80 kW ancillary ac generators.

(3) Adequacy of the 60 amps breaker for full load testing of the 80 kW ancillary ac generators.

In a letter dated August 5, 2010, the applicant stated that it had reviewed the adequacy and accuracy of both the rating of the distribution panel and size of each of the breakers on the ancillary diesel generator bus. Based on its review, the applicant revised the distribution panel rating and breaker sizes as follows:

The 125 amp breaker to protect the 45 kVA regulating transformer will be changed from 125 amp breaker to 20 amp breaker, as was originally designed, to a breaker size appropriate to the load on the transformer under ancillary diesel generator operating conditions of 20 amps.

(1) The distribution panel rating will be increased from 100 amps to 225 amps so that it will be protected by the generator output breaker. This generator output breaker will be changed from 100 amps to 150 amps so as to be greater than 125 percent of the generator output.

(2) The 60 amp breaker for full load testing of the 80 kW ancillary generators will be replaced with a 150 amp breaker to allow for full testing of the ancillary diesel generator.

(3) The 50 amp breaker to protect PCS motor will be changed to 100 amps in order to avoid spurious tripping of the breaker on PCS motor start.

The staff has reviewed the above information and concludes that the proposed revised rating of the distribution panel and the size of each of the breakers on the ancillary diesel generator bus are compatible with the revised rating of the ancillary diesel generator. Therefore, the staff finds this concern resolved.

23.D.4 Conclusion

The staff's review concludes that the proposed design changes above are acceptable since the proposed changes will not adversely affect safety-related SSCs and the capability of the ancillary diesel generators to perform their post-72 hour function. These design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable. RAI-DCP-CN-59-SBP-01 and RAI-SRP 8.3.1-EEB-2 are resolved.

The staff has reviewed the proposed changes to the ancillary diesel generator design and concludes that the ancillary diesel generator distribution panel and the associated load feeder breakers are sized in accordance with the National Electric Code and, therefore, the proposed changes are acceptable. Also, the staff finds the changes made in Figure 8.3.2-2 to be minor and acceptable.

23.E Changes to Potable Water System

23.E.1 Description of Proposed Changes

In letters dated April 26, 2010, and August 2, 2010, The applicant proposed changes to modify the design of the potable water system (PWS) to add a safety-related loop seal in the PWS piping that penetrates the MCR envelope boundary to prevent in-leakage into the MCR envelope during MCR emergency habitability system (VES) operation.

23.E.2 Regulatory Basis

The applicable regulatory requirement and regulatory guidance are as follows:

- 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 2, "Design Basis for Protection against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- Regulatory guide (RG) 1.29, "Seismic Design Classification"

23.E.3 Evaluation

The current PWS is a nonsafety system. The applicant proposed a design modification of the PWS for the consideration of control room envelope integrity under a seismic event. If a pipe break occurs resulting from a seismic event, the current design of the PWS would not maintain the MCR VES pressure boundary. In the proposed design modification, the applicant added an isolation valve, a loop seal, and a vacuum breaker to upgrade portions of the PWS to provide safety-related function for MCR isolation in meeting RG 1.197 for MCR envelope integrity.

To meet the requirements of GDC 2 as it relates to structures and systems being capable of withstanding the effects of natural phenomena including seismic event, acceptance depends on meeting the guidance of the portions of Regulatory Position C.1 of RG 1.29 for the safety-related portions of the system and Regulatory Position C.2 of RG 1.29 for the nonsafety-related portions of the system.

To determine the adequacy of the safety-related function of the system modification, the staff requested more detailed information in RAI-DCP-CN01-SBP-COLP-01. In the RAI, the staff requested the applicant provide additional information pertaining to: (1) a figure and system description for the illustration of the system modification in the DCD; (2) the design basis and maintenance requirements for the loop seal; (3) the DCD changes including both Tier 1 and Tier 2; (4) the single failure discussion for the vacuum breaker; (5) the minimum response time for operator manual actions; (6) flood protection; (7) testing requirements; (8) TS requirements; and (9) MCR envelope integrity.

In a letter dated August 2, 2010, responding to RAI-DCP-CN01-SBP-COLP-01, the applicant provided more detailed information. Based on the review of the additional information, the staff found the following discussion from the RAI responses to form the basis for its finding:

1. Figure 9.2.5-1, “Main Control Room Potable Water System Isolation,” and system description are added in the DCD Tier 1 and Tier 2 to illustrate the portions of the PWS that have safety-related function pertaining to preventing in-leakage into the MCR envelope.

2. The loop seal is designed to maintain MCR isolation providing the operator sufficient time to close the PWS isolation valves in the event that the nonseismic PWS piping broke. The potable water tank is located at a higher elevation that maintains water in the loop seal by design. Therefore, no maintenance requirements are needed for the loop seal.

3. DCD changes were made in Tier 1 Section 2.7.1, Table 2.7.1-1 and Table 2.7.1-2, to include the safety-related portions of the PSW. These two tables are referenced by ITAAC Table 2.7.1-4. In Tier 2, changes were made in Table 3.2-3, Table 3.9-12, Table 3.9-16, Table 3.11-1, Section 9.2.5.1.1, Section 9.2.5.4, and Section 14.2.9.1.6, which address the safety design basis, safety-related portions of the PWS in AP1000 Class C, seismic Category I, ASME III-3, inservice test requirements, and environmental qualification.

4. The vacuum breaker is to help ensure that a break in the nonsafety-related potable water supply piping would not cause water to siphon from the loop seal. These pressure/vacuum relief devices are not required to consider single active failures. This is consistent with the implementation of a single vacuum relief device in the automatic depressurization system and VES.

5. The safety-related portions of the potable water seal assure MCR pressure boundary integrity after a design basis event. Since a seismic event is not assumed in the analysis to occur simultaneously with another design basis event (such as loss-of-coolant accident (LOCA)), there are no radiological conditions that would change MCR habitability. VES actuation occurs when there is sustained loss of electrical power. For this situation, the design basis for VES actuation is to provide breathable air for the MCR occupants. The MCR occupants will have a supply of breathable air for 72 hours and the MCR would remain habitable. The operators would be alerted to a loss of air through the loop seal piping by the low differential pressure alarm for the MCR and would close the potable water manual isolation valves. The time for the operator actions is not a critical parameter for the safety-related function.

6. The nonseismic PWS lines in the MCR are limited to the kitchen and restroom areas with line sizes of 2.5 cm (1 in) and smaller. There are no safety-related components that would be adversely affected by a rupture of the nonseismic PWS piping in the MCR kitchen and bathroom areas.

7. The initial test program and inservice testing included the PWS to ensure the integrity of the MCR pressure boundary; as shown in Tier 2 Section 14.2.9.1.6, Table 3.9.6, and Table 3.9-16.

8. TS, Section 3.7.6, “Main Control Room, Habitability System,” has been updated to include the safety-related isolation valves of the PWS. The MCR pressure boundary includes the PWS water seal that prevents gas flow through the piping. These TS changes were addressed in the response to RAI-SRP-6.4-SPCV-03.

9. The sanitary drainage system (SDS) is one of the systems that penetrate the MCR boundary. Portions of the SDS, including isolation valves and loop seals, have been modified to safety-related and seismic Category I. The modification of the PWS, DCD Tier 1 and Tier 2 information pertaining to the SDS was revised accordingly. The integrity of the AP1000 MCR boundary was addressed in the response to RAI-SRP-6.4-SPCV-03.

The staff has determined that the proposed modification meets RG 1.29, Position C.2 for the nonsafety-related PSW based on the information in Items (1) and (6) above – that the change portions of the nonsafety-related PWS will be safety-related, and that a failure of the nonsafety-related portions of the system will not adversely affect any of the safety-related functions. Further, the DCD information discussed in Item (3) above demonstrates that RG 1.29, Position C.1 is met for the safety-related portions of the system because appropriate classification designations are specified for the PWS consistent with the approach described in DCD Tier 2, Section 3.2. Based on meeting the guidance of RG 1.29, Positions C.1 and C.2, the staff has determined that the modified PWS meets GDC 2.

The staff reviewed the response to Items (2) and (4) and agrees with the applicant’s justifications that: (1) no maintenance is required for the loop seal because the elevation of the water tank is sufficient to provide the water seal without maintenance; and (2) the vacuum breaker is not required to consider single active failures because it is consistent with the VES design for the automatic pressure relieve devices.

The staff reviewed the response to Item (5) for the human factors concern of the minimum response time for operator manual actions. The applicant explained that operator action was not credited for any design basis event. The safety-related (seismic) design of the loop seal and surrounding pipe in coordination with other safety-related systems used to manage design basis events ensure that either the hazardous environment that would necessitate control room isolation is prevented and/or that the control room isolation function is maintained. The manual valves were installed as a defense-in-depth measure to address longer-term evaporation of the water in the loop seal, and for deterministic beyond-design-basis evaluations that simply considered the loop seal to be unavailable. There is no regulation or regulatory guidance applicable to operator manual actions used in this manner.

In reviewing Items (7) through (9), the staff finds that there is sufficient information to address the need for testing the isolation valves and establishing their TS to ensure the integrity of the MCR pressure boundary. The review of the integrity of the MCR pressure boundary is under

NUREG-0800 Section 6.4. The proposed changes were found acceptable in SER Section 6.4.1.3 because the overall effectiveness of the control room envelope is demonstrated through the testing associated with the control room integrity program.

Based on the above, the staff has determined that sufficient information is provided to address the staff's questions posed in RAI-DCP-CN01-SBP-COLP-01 and that sufficient details are provided in the DCD markups to address the safety-related function of the PWS. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.E.4 Conclusion

The staff's review concludes that the design changes described above are acceptable because they meet the requirements and guidance of GDC 2, RG 1.29. RAI-DCP-CN01-SBP-COLP-01 is resolved.

23.F Changes to Reactor Coolant Pressure Boundary Leakage Detection

23.F.1 Description of Proposed Changes

In letters dated January 20, 2010, and July 29, 2010, the applicant proposed changes to revise the licensing basis for unidentified RCS leak detection by removing the N13/F18 containment atmosphere radiation monitor and replacing it with the Flourine-18 (F-18) particulate monitor. The TS Limiting Condition for Operation (LCO) 3.4.9a, and the related discussions in TS Bases B 3.4.7 and B 3.4.9, are revised to reflect the change from the "N13/F18 gaseous monitor" to an "F18 particulate monitor."

In the process of reviewing this proposed design change, the staff identified an operating issue pertaining to RCS leakage detection. Operating experiences at Davis Besse (NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity") indicated that prolonged low-level unidentified reactor coolant leakage inside containment could cause material degradation that could compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary.

23.F.2 Regulatory Basis

The applicable regulatory requirement and regulatory guidance are as follows:

- GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to providing means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage

- 10 CFR 52.79(a)37, "Contents of applications; technical information in final safety analysis report," as it relates to "information necessary to demonstrate how operating experience insights have been incorporated into the plant design"

- RG 1.45, Revision 1, as it relates to "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the containment atmosphere radioactivity monitoring system design information as described in DCD Sections 5.2.5 and 11.5. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified in NUREG-0800 Chapter 16.

23.F.3 Evaluation

GDC 30 requires that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. RG 1.45 describes acceptable methods for implementing GDC 30.

<u>F18 Particulate Monitor</u>

The staff reviewed the proposed design changes, including changes in DCD Tier 2, Appendix 1A, Sections 3.1.4, 3.6.3.3, 5.2.5.3.3, and 11.5.2.3.1, and TS LCO 3.4.9, Sections B.3.4.7 and 3.4.9. The proposed design changes revise the licensing basis for unidentified RCS leak detection by removing the N13/F18 containment atmosphere radiation monitor and replacing it with an F-18 particulate monitor.

<u>Detector Sensitivity</u>

In the review, the staff determined that the applicant did not provide sufficient information to demonstrate that the newly proposed F-18 particulate radiation monitor (PSS-JE-RE027) sensitivity is capable of detecting the RCS leak rate of 1.9 lpm (0.5 gpm) according to the TS. Therefore, in RAI-SRP11.5-CHPB-05, the staff requested additional information on the analysis demonstrating the sensitivity of the proposed radiation monitor.

Although this design change specifies the radiation monitor sensitivity for particulate radioactivity, the change does not provide an analysis to demonstrate that the specified monitor sensitivity is capable of satisfying the technical basis of using realistic radioactive concentrations in the RCS, as described in RG 1.45, Revision 1.

In a May 14, 2010, response to RAI-SRP11.5-CHPB-05, the applicant provided the analysis demonstrating that the monitor can detect a RCS leak rate [ ] lower than that specified in the AP1000 TS. Based on its review, the staff found that the results of the applicant's analysis were highly and directly dependent upon the F-18 concentration in the reactor coolant and the fraction of F-18 assumed to enter the containment atmosphere.

The applicant derived the coolant concentration from two references [ ] and to be conservative and to minimize the amount of radioactivity leaked from the RCS, the applicant used the lower value in its analysis. The staff's own literature review of F-18 in pressurized-water reactor (PWR) reactor coolant indicated that the concentration could be 0.014 uCi/ml, [ ] which is lower than that used by the applicant.

The applicant estimated the amount of F-18 entering the containment atmosphere to be [ ] a small fraction of the total leaked from the RCS. The applicant derived this fraction from an analysis that estimated the fraction flashed into the space between a RCS coolant pipe and the insulation surrounding the pipe, and the fraction escaping from the insulation into the containment atmosphere. The applicant estimated the fraction flashed into the space between the pipe and insulation using a method described in NUREG-1320, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," dated May 1988. Using an aerosol transport code, the applicant then calculated the fraction escaping the insulation into the containment atmosphere. The staff independently verified the flash fraction calculation using the method described in NUREG-1320. The staff also concluded that the final escape fraction calculated by the aerosol transport model was conservative because almost all aerosols larger than 1 micron diameter fail to escape into the containment atmosphere, thus greatly reducing the amount of radioactivity reaching the monitor.

Based on its review and independent verification, the staff concludes that the proposed monitor is sufficiently sensitive to detect the TS leak rate. Hence, the staff closed RAI-SRP11.5-CHPB-05.

Response Time

In the letter dated March 12, 2010, the applicant proposed to change the radiation monitor response time for the leak detection from 1.9 Lpm (0.5 gpm) within one hour to 1.9 Lpm (0.5 gpm) within two hours. However, in a May 14, 2010, response to RAI-SRP11.5-CHPB-05, the applicant stated that the radiation particulate monitor is capable of detecting a 1.9 Lpm (0.5 gpm) leak in one hour, and the applicant provided corresponding Tier 2 DCD changes. In a teleconference on July 29, 2010, the applicant clarified that the letter dated May 14, 2010, superseded the early letter regarding the response time. The staff has determined that the response time of detecting 1.9 Lpm (0.5 gpm) leakage in one hour as specified in the May 14, 2010, letter is consistent with the certified design of DCD Revision 15. Based on the clarification and the updated DCD Tier 2 changes, the staff has determined that no change in the response time for the radiation monitor is necessary and, therefore, detecting 1.9 Lpm (0.5 gpm) leakage within one hour is acceptable.

Based on the above, the staff has determined that the proposed F-18 particulate radiation monitor sensitivity and response time pertaining to the RCS detection function are acceptable. Further, using a particulate radiation monitor as one of the leakage detection instruments is consistent with the guidance in RG 1.45, Revision 1. Therefore, the staff has determined that the proposed F-18 particulate radiation monitor is acceptable.

With respect to proposed changes to GTS 3.4.9 and their associated bases, the staff finds these changes, as modified by the applicant's response to RAI-SRP11.5-CHPB-05 acceptable because they reflect the system design and operating information described in DCD Sections 5.2.5 and 11.5.

Davis Besse Operating Experience with RCS Leakage Detection

NRC Bulletin 2002-01 described that the operating experience at Davis Besse in 2002 indicated that prolonged low-level unidentified reactor coolant leakage resulting from nozzle cracking of the control rod drive mechanism inside containment could cause material degradation that could compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. The question was raised regarding licensees' practices for identifying and resolving degradation of the reactor coolant pressure boundary. Pursuant to 10 CFR 52.79(a)37, relating to the requirement to provide "information necessary to demonstrate how operating experience insights have been incorporated into the plant design," in RAI-DCP-CN45-SBP-01, the applicant was requested to address this issue.

<u>COL Information Item</u>

In a letter dated July 29, 2010, responding to RAI-DCP-CN45-SBP-01, the applicant revised DCD to add a new COL information item, COL Information Item 5.2-3, and a new Section 5.2.6.3 to describe COL Information Item 5.2-3 as follows:

> 5.2.6.3 Response to Unidentified Reactor Coolant System Leakage Inside Containment
>
> The Combined License applicant will provide information to address prolonged low-level unidentified reactor coolant leakage inside containment which could cause material degradation such that it could potentially compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. This issue could be addressed by operating procedures. The procedures should address operator actions in response to prolonged low level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide operator sufficient time to take actions before the TS limit is reached. The procedures should address identifying, monitoring, trending, and repairing prolonged low-level leakage. The procedures should also define the alarm setpoints and demonstrate that the setpoints are sufficiently low to provide an early warning for operator actions prior to Technical Specification limits. In addition, the procedures should address converting the instrument output to a common leakage rate.

The staff reviewed the RAI-DCP-CN45-SBP-01 response and determined it to be acceptable because the description of the COL information item in DCD Section 5.2.6.3 markups is consistent with the guidance in RG 1.45, Revision 1, pertaining to managing the prolonged low-level reactor coolant system leakage. Therefore, GDC 30 is met based on the conformance to RG 1.45. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.F.4 Conclusion

The staff's review concludes that the design changes described above are acceptable because they meet the requirements and guidance of GDC 30 and RG 1.45, Revision 1. In addition, this design change was evaluated with respect to the adequacy of TS that have been established for the RCS leak detection instrumentation system and found acceptable. COL applications that incorporate by reference the DCD must address the new COL information item described in DCD Section 5.2.6.3. RAI-SRP11.5-CHPB-05 and RAI-DCP-CN45-SBP-01 are resolved.

23.G Changes to Spent Fuel Flood-up Valves Remote Position Indication

23.G.1 Description of Proposed Changes

In letters dated April 26, 2010, and August 3, 2010, the applicant proposed changes to the spent fuel flood-up valves remote position indication. AP1000 DCD, Revision 17 requires several valves that connect to the refueling cavity to have their position status monitored during plant shutdowns to prevent draining of the refueling cavity and the SFP. Tier 1 Table 2.3.7-1 presents a list of these valves. These valves are also designed as seismic Category I components and identified as such in Tier 1 of the DCD. The applicant has identified in these proposed design changes that the spent fuel pool cooling system (SFS) valves SFS-PL-V031 and SFS-PL-V033 are also required to have their position status monitored during plant shutdowns to prevent draining of the refueling cavity and the SFP, but were not identified as such in the DCD. This design change proposes to modify Tier 1 Table 2.3.7-1 to require that these valves have their position status displayed in the MCR. In addition, a previous design change identified that valve SFS-PL-V075 is required to be locked open during normal operation to provide a flow path during scenarios requiring containment flood-up. During refueling, V075 provides the same function of preventing the draining of the refueling cavity and the SFP as V031 and V033. This design change proposes to modify Tier 1 Table 2.3.7-1 to require valve V075 to have its position indicated in the MCR as well. This design change proposes to add two external Class 1E limit switches (open/closed) to the following isolation valves:

- SFS refueling cavity drain to steam generator system (SGS) compartment isolation valve (SFS-PL-V031)

- SFS refueling cavity drain to compartment sump isolation valve (SFS-PL-V033)

- SFS containment floodup isolation valve (SFS-PL-V075)

23.G.2 Regulatory Basis

The regulatory basis for evaluating these proposed design changes are documented in Section 7.5 and Section 9.1.2 of NUREG-1793. In particular, GDC 61, “Fuel Storage and Handling and Radioactivity Control,” requires the fuel storage system to be designed for adequate safety under anticipated operating and accident conditions. The system must be designed with the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions. During refueling operations the reactor cavity is connected to the SFP, and failure of the valves and piping sections identified above could drain the water from the refueling cavity and from the SFP. Acceptability of the proposed design changes was based on conformance with the existing AP1000 licensing basis and the guidance specified in NUREG-0800 Section 9.1.2 (as applicable).

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h), “Codes and standards,” and 10 CFR 52.47, “Contents of applications; technical information.” The changes must also conform to the requirements of GDC 13, “Instrumentation and Control,” and GDC 19, “Control Room,” in 10 CFR Part 50, Appendix A, and should meet guidance in RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.” The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.5, “Information Systems Important to Safety.” Acceptability was also based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Section 7.5 (as applicable).

23.G.3 Evaluation

During the staff’s evaluation of these proposed design changes, it was determined that additional information was required from the applicant. The staff identified that the markup of DCD pages affected by the proposed design changes included changes that were not directly related to the justification provided in the design change submittal. In RAI-DCP-CN55-SBP-01.a and RAI-DCP-CN55-SBP-01.b, the applicant was requested to provide justification for these additional changes. In the RAI response letter dated August 3, 2010, the applicant stated that the changes identified by the staff in RAI-DCP-CN55-SBP-01.a were not part of this proposed design change, but were part of RAI-SRP6.4-SPCV-03, R2. The staff found the applicant's response acceptable because these changes were already evaluated by the staff in Section 6.4 of this report. Therefore, RAI-DCP-CN55-SBP-01.a is considered closed.

In response to RAI-DCP-CN55-SBP-01.b, the applicant stated that the deletions in Tier 2 Table 3.9-16 identified by the staff in RAI-DCP-CN55-SBP-01.b were not deleted from the table, but were moved to the next page. Since this represents an editorial change, the staff finds the applicant's response acceptable and RAI-DCP-CN55-SBP-01.b is considered closed.

In RAI-DCP-CN55-SBP-01.c, the staff identified that the Tier 1 drawings depicting this system did not show all the valves mentioned in the design change proposal (DCP). Therefore, the applicant was requested to update the Tier 1 drawings impacted by this design change. In the RAI response letter dated August 3, 2010, the applicant included a markup of Tier 1 Figure 2.3.7-1, which now included valve SFS-PL-V075. Therefore, the staff finds the applicant's response to be acceptable and RAI-DCP-CN55-SBP-01.c is considered closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

DCD Tier 2 Section 9.1.3.3.5, "Spent Fuel Pool Cooling System Valves," provides a description of the valve arrangement needed for refueling. It was not clear to the staff if this description was impacted by the proposed change. In RAI-DCP-CN55-SBP-01.d, the applicant was requested to confirm that the configuration description provided in Tier 2 Section 9.1.3.3.5 was still valid and had not been impacted by the proposed change. In the RAI response letter dated August 3, 2010, the applicant stated that the valve configuration description provided in Tier 2 Section 9.1.3.3.5 is still valid and has not been impacted by the proposed change. The staff agrees with the applicant's determination that Tier 2 Section 9.1.3.3.5 is still valid; therefore, RAI-DCP-CN55-SBP-01.d is considered closed.

The staff evaluated Tier 2 Figure 9.1-6 (Sheet 1 of 2) "Spent Fuel Pool Cooling System Piping and Instrumentation Diagram," and identified additional possible refueling cavity drain paths. In RAI-DCP-CN55-SBP-01.e, the applicant was requested to confirm that all refueling cavity drain path isolation boundary piping and components were described in Tier 2 and included in the appropriate Tier 1 sections and tables. In the RAI response letter dated August 3, 2010, the applicant stated that all refueling cavity penetrations and associated isolation valves that are at elevations below the minimum safety level outlined in Tier 2, Chapter 16, TS 3.9.4 are designed as seismic Category I, and have been identified in the appropriate DCD (Tier 1 and Tier 2) section. The applicant also clarified that all refueling cavity penetration lines and associated isolation valves that are not seismic Category I are located at elevations that preclude the possibility of draining the refueling cavity below the minimum safe level for refueling operations. The staff finds that the applicant has properly identified all the possible refueling cavity drain paths in the appropriate DCD sections, and that these drain paths are designed as seismic Category I components. Therefore, RAI-DCP-CN55-SBP-01.e is considered closed.

As documented in NUREG-1793, the staff reviewed and approved the SFS in AP1000 DCD Tier 1 Section 2.3.7 and Tier 2 Section 9.1.3. The staff reviewed the applicant's proposed design changes to the SFS in the AP1000 standard design by using the review procedures described in NUREG-0800 Section 7.5 and requirements of GDC 13 and GDC 19 in 10 CFR Part 50, Appendix A. AP1000 DCD, Revision 17, Tier 1 Table 2.3.7-1 shows that the safety-related display in the MCR is required for the SFS refueling cavity drain to the SGS compartment isolation valve (SFS-PL-V031) and SFS refueling cavity drain to the compartment sump isolation valve (SFS-PL-V033). These two valves are also required to have their position status monitored during plant shutdowns to prevent draining of the SFP. However, in AP1000 DCD, Revision 17, these two valves were designed without limit switches. To achieve the above required safety-related display functions, two external Class 1E limit switches (open/closed) are provided for valve (SFS-PL-V031) and valve (SFS-PL-V033).

The SFS containment floodup isolation valve (SFS-PL-V075) was added to the SFS by a previous DCP. This valve is required to be locked open to provide a flow path during scenarios requiring containment flood-up. Valve SFS-PL-V075, when closed, also provides the function of preventing the refueling cavity from draining. This function requires the status indication displayed for this valve in the MCR. But, the current valve functional requirements do not provide for remote position indication. Therefore, two external Class 1E limit switches (open/closed) have to be provided for the SFS containment floodup isolation valve to achieve the required safety-related display function.

For the specific DCD changes, the applicant added valve SFS-PL-V075 to DCD Tier 1 ITAAC Table 2.3.7-1 and the new SFS containment flood-up line to DCD Tier 1 ITAAC Table 2.3.7-2. The applicant also added the status of the above three valves to DCD Tier 2, Table 3.9-16, “Valve Inservice Test Requirements”; Table 3.11-1, “Environmentally Qualified Electrical and Mechanical Equipment”; Table 3.2-3, “AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment”; Table 7.5-1, “Post Accident Monitoring System”; and Table 7.5-7, “Summary of Type D Variables.” The staff finds that the proposed design changes described above are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.G.4 Conclusion

The staff evaluated these proposed design changes against conformance with the existing AP1000 licensing basis and the guidance specified in NUREG-0800 Sections 7.5 and 9.1.2 (as applicable). The staff also evaluated these proposed design changes against the requirements of GDC 61, which requires the fuel storage system to be designed for adequate safety under anticipated operating and accident conditions.

The staff concludes that, based on the description provided in these design changes and the RAI responses discussed above, these proposed design changes are in compliance with the requirements of GDC 61 and follow the guidance provided in NUREG-0800 Sections 9.1.2 and 7.5; therefore, the staff finds the proposed changes to be acceptable.

23.H Changes to the AP1000 Steam Generator Thermal-Hydraulic Data Report

23.H.1 Description of Proposed Changes

In letters dated May 10, 2010; August 4, 2010; and August 12, 2010; the applicant proposed the following changes to the AP1000 steam generator (SG) design:

- In DCD Section 5.4.4.3, the pressure drop through the SG flow restrictor at 100 percent steam flow is changed from approximately 103 kiloPascal (kPa) (15 pounds per square inch (psi)) to approximately 138 kPa (20 psi).

- In DCD Table 5.4-5, the SG design fouling factor is changed from 1.937×10^{-5} to 1.586×10^{-5} m^2-°C/watt (W) (1.1×10^{-4} to 9.0×10^{-5} hr-°F-ft^2/British Thermal Unit (BTU)).

- In DCD Table 10.3.2-2 (and TS Table 3.7.1-2), the main steam safety valve (MSSV) lifting settings and relieving capacities are changed as follows:

Valve Nos.		Lift Setting kPa (psig)		Relieving Capacity 10^6 kg/hr (lb/hr)	
		From	To	From	To
MSSV #1	V030A/B	8170 (1185)	8170 (1185)	0.5940 (1.310)	0.5990 (1.320)
MSSV #2	V031A/B	8246 (1196)	8253 (1197)	0.5987 (1.320)	0.6078 (1.340)
MSSV #3	V032A/B	8329 (1208)	8336 (1209)	0.6105 (1.346)	0.6123 (1.350)
MSSV #4	V033A/B	8405 (1219)	8418 (1221)	0.6151 (1.356)	0.6169 (1.360)
MSSV #5	V034A/B	8487 (1231)	8494 (1232)	0.6205 (1.368)	0.6214 (1.370)
MSSV #6	V035A/B	8563 (1242)	8494 (1232)	0.6214 (1.370)	0.6214 (1.370)

The proposed changes to the MSSV lift settings and relieving capacities are in response to an increased pressure drop calculation for the SG steam outlet nozzle, which impacts the MSSV inlet line losses.

23.H.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes to lift settings and relieving capacity is documented in Chapters 10 and 15 of NUREG-1793. GDC 10, “Reactor Design,” and GDC 15, “Reactor Coolant System Design,” respectively, in 10 CFR Part 50, Appendix A, require that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs). The design basis events in DCD Chapter 15 were analyzed to ensure compliance with GDC 10 and GDC 15.

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff’s evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the MSSV design and operating information described in DCD Section 10.3. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, “Technical Specifications – Revision 3, March 2010.” Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.H.3 Evaluation

As discussed in the August 12, 2010, response to RAI-DCP-CN58-SRSB-02, the applicant revised the calculation of the pressure differential across the SG steam outlet nozzle flow restrictor in the thermal-hydraulic data report, which is a document typically provided by the manufacturer to the utility operating the SG. The revised calculation showed an increase of the pressure loss from approximately 103 kPa (15 psi) to 138 kPa (20 psi). Therefore, the applicant proposed to revise DCD Section 5.4.4.3 by stating that the resultant pressure drop through the SG flow restrictor at 100 percent steam flow is approximately 138 kPa (20 psi), rather than approximately 103 kPa (15 psi) in DCD Revision 17. Also, the SG design fouling factor is revised from 1.937×10^{-5} to 1.586×10^{-5} m^2-°C/W (1.1×10^{-4} to 0.9×10^{-5} hr-°F-ft^2/BTU). The applicant stated that this reduction is based on the operating experience of replacement SGs with Alloy 690 tubing since 1989, and SG fouling factor is reduced to offset the increased pressure loss while still maintaining sufficient margin for the SG heat transfer performance. For the Chapter 15 safety analysis, a higher fouling factor was used where reduced heat transfer to the SG is limiting. The proposed lower value of SG design fouling factor provides increased margin. Therefore, there is no effect on the safety analysis.

Because of the increased pressure loss through the SG flow restrictor, the applicant proposed to change the relief capacities and lift settings of the MSSVs in accordance with ASME Code Section III, Subsection NC-7300. The setpoint of MSSV #6 (V035A/B), which is the highest MSSV setpoint, was lowered to account for the increased pressure losses and still maintain the required relieving capacities. In response to RAI-DCP-CN58-SRSB-01, the applicant stated that for the purposes of the AP1000 safety analysis, the significant MSSV setpoints are MSSV #1 (V030A/B) and MSSV #6 (V035A/B). MSSV #1 represents the lowest safety valve setpoint and is used to determine if design transients will challenge the MSSVs. MSSV #6 is the highest valve setpoint and is used to determine the overall steam pressure relief capacity in the safety analysis. In this proposed change, the lift setpoint of MSSV #6 is reduced from 8563 kPa (1242 psig) to 8494 kPa (1232 psig) to account for the impact of the revised SG flow restrictor pressure loss, but the relieving capacity of MSSV remains unchanged. The reduced lift setting resulted in a marginal increase in valve capacity with respect to the safety analysis of the limiting overpressure turbine trip event. Also, the lift setting of MSSV #1 is not changed while its relieving capacity is increased slightly. The setpoints and relieving capacities of MSSV #2 through MSSV #5 are also slightly increased, but they do not affect the safety analyses. Therefore, the proposed slight changes to the MSSV setpoints and relieving capacities have minimal effect on the safety analyses of the overpressure events. For the limiting turbine trip event, the minimum departure from nucleate boiling ratio (DNBR) remains well above the safety analysis DNBR limit, and the RCS pressure and the secondary side steam pressure remain within 110 percent of the respective design pressures.

With respect to proposed changes to TS 3.7.1, the staff finds these changes acceptable because they reflect the MSSV design and operating information described in DCD Section 10.3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text.

23.H.4 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes are acceptable because these changes have minimal effect on the safety analysis, and GDC 10 and GDC 15 continue to be complied with. These design changes were also evaluated with respect to the adequacy of TS that have been established for the MSSVs. The staff finds these changes acceptable.

23.I Changes Related to the Implementation of P-17 for Rod Withdrawal Prohibit

23.I.1 Description of Proposed Changes

In a letter dated May 10, 2010, the applicant proposed design changes related to the implementation of P-17 logic for rod withdrawal prohibit. The current design requires the P-17 signal coincident with the Beacon Unavailable Signal to generate the automatic rod withdrawal prohibit. The proposed changes would remove the Beacon Unavailable Signal and the associated AND logic gate to enable an automatic rod withdrawal prohibit solely on the rate of change in nuclear power (P-17). The implementation of the P-17 logic to prohibit rod withdrawal is a conservative change from the current design. As a result of this proposed P-17 logic change, the applicant proposed to revise DCD Section 15.4.3.2 by revising the sequence of dropped rod event in DCD Table 15.4-1 and DCD Figures 15.4-1 through 15.4-4 to reflect this change.

23.I.2 Regulatory Basis

The regulatory basis for evaluating the changes related to the implementation of P-17 logic for rod withdrawal prohibit is documented in Chapter 15 of NUREG-1793. The proposed design changes must also conform to the requirements of GDC 10, GDC 13, GDC 20, "Protection System Functions," and GDC 25, "Protection System Requirements for Reactivity Control Malfunctions" in 10 CFR Part 50, Appendix A.

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that SAFDLs are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

GDC 25 requires that the reactor protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

23.I.3 Evaluation

The proposed changes to enable the P-17 automatic rod withdrawal prohibit solely on the rate of change in nuclear power affect the response to dropped rod control cluster assembly (RCCA) events, described in DCD Section 15.4.3.2.1. A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the power distribution. In the automatic control mode, the plant control system detects the drop in the core power and initiates withdrawal of a control bank, which could result in power overshoot. The implementation of the proposed P-17 logic to prohibit automatic rod withdrawal prevents the potential power overshoot and is, therefore, conservative relative to the current design. Therefore, the safety analysis of dropped RCCA events would be bounded by the existing analysis described in DCD Section 15.4.3.2.1, which demonstrated that the minimum DNBR for one or multiple rod drops from the same group is greater than the DNBR limit.

The applicant has run a spectrum of transients, varying key input parameters to verify that DNBR limits continue to be met with the proposed P-17 logic implementation. The sequence of a rod drop event displayed in DCD Table 15.4-1 is considered to be representative of all of the cases run. The results of a representative dropped RCCA event are provided in revised Figures 15.4.3-1 through 15.4.3-4. DCD Table 15.4-1 is also modified to reflect a representative case where the peak nuclear power occurs at time 21.7 seconds and peak core heat flux occurs at 24.2 seconds.

The staff has reviewed the proposed design changes to implement the P-17 logic for rod withdrawal prohibit. Based on its review, the staff finds that this change does not affect the safety analysis of the dropped RCCA events described in DCD Section 15.4.3.2.1, and the analysis continues to satisfy the acceptance criteria of NUREG-0800 Section 15.4.3 with respect to the minimum DNBR, peak pressure, and fuel cladding integrity. Therefore, the staff concludes that the proposed design changes are acceptable.

23.I.4 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 proposed design changes are acceptable because they meet the requirements of GDC 10, GDC 13, GDC 20, and GDC 25.

23.J Changes Related to Post-Design Basis Accident Transmitters

23.J.1 Description of Proposed Changes

In letters dated May 25, 2010, July 29, 2010, and October 20, 2010, the applicant proposed design changes to relocate seven containment pressure transmitters outside containment and connect to remote pressure sensors inside containment by sealed capillary tubing. These changes require the addition of four new containment penetrations, one for each safety division. The applicant also proposed to relocate 18 Category 1 post accident monitoring system (PAMS) transmitters above the maximum design-basis accident (DBA) flood level. In addition, the applicant proposed to reduce post-accident operability time for 18 Category 2 PAMS transmitters from 4 months to 2 weeks.

23.J.2 Regulatory Basis

The regulatory basis for evaluating the changes to post-DBA transmitter requirements is documented in Chapters 3, 5, 6, 7, and 9 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of GDC 13 in 10 CFR Part 50, Appendix A, and should meet the guidance in RG 1.11, “Instrument Lines Penetrating Primary Reactor Containment”; RG 1.97, and RG 1.151, “Instrument Sensing Lines.” The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.5, “Information Systems Important to Safety.” Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by NUREG-0800 Section 7.5 and RG 1.11, RG 1.97, and RG 1.151 (as applicable). The addition of four new containment penetrations was evaluated using the guidance provided in NUREG-0800

Section 6.2.4, “Containment Isolation System”; NUREG-0800 Section 6.2.6, “Containment Leakage Testing”; and the requirements of 10 CFR Part 50, Appendix J, “Containment Leak Rate Testing,” and GDC 52, “Capability for Containment Leak Rate Testing.”

23.J.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the post-DBA transmitter requirements in AP1000 DCD, Revision 15. The staff reviewed the applicant’s proposed design changes using the review procedures described in NUREG-0800 Sections 6.2.4, 6.2.6, and 7.5, requirements of GDC 13 and GDC 52 in 10 CFR Part 50, Appendix A; 10 CFR Part 50, Appendix J; and the guidance in RG 1.11, RG 1.97, and RG 1.151.

The applicant proposed to relocate seven containment pressure transmitters outside containment. These changes will allow direct measurement of differential pressure across the containment shell. These changes also allow those seven transmitters to be located in a mild environment. Relocation of the seven transmitters outside of containment could address the LCO in TS that containment pressure shall be maintained between -1.38 kPa (-0.2 psig) to 6.9 kPa (+1.0 psig), which are used in safety analysis.
[

] Relocation of the seven transmitters outside containment allows the plant to operate within TS for containment pressure. Moving the transmitters outside containment also eliminates the need to include a DBA environmental allowance in the determination of channel accuracy and the setpoint for PCS actuation.

Moving seven containment pressure transmitters outside containment and connecting to remote pressure sensors inside containment by sealed capillary tubing requires the addition of four new containment penetrations, one for each safety division. In Divisions A, B and C, the normal-range and wide-range transmitters share one capillary line and penetration. For Division D there is no wide-range transmitter, so the normal-range transmitter is on its own capillary. These four new instrument penetrations, P46, P47, P48, and P46 are identified in marked up DCD Tier 1 Table 2.2.1-1 and Figure 2.2.2-1, “Containment System.” In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

GDC 55, “Reactor Coolant Pressure Boundary Penetrating Containment,” or GDC 56, “Primary Containment Isolation,” usually require each line that penetrates the containment and is either a part of the reactor coolant pressure boundary or that connects directly to the containment atmosphere, to be provided with containment isolation valves. However, NUREG-0800 Section 6.2.4 endorses the containment isolation provisions described in RG 1.11 for instrument sensing lines. RG 1.11 finds sensing lines with no isolation valves acceptable as long as the lines are sized to limit the potential offsite exposure to be below the guidelines of 10 CFR Part 100, “Reactor site criteria.” The capillary tubing meets this criterion.

The four new containment penetrations to accommodate the new instrument capillary tubing will be leak rate tested, as required by GDC 52. The four new penetrations will be Type A tested, which meets the requirements of 10 CFR Part 50, Appendix J and the guidance of NUREG-0800 Section 6.2.6. This leak rate testing commitment will be added to DCD Tier 2, Table 6.2.3-1, “Containment Penetrations and Isolation Valves.” In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The applicant also proposed to move 18 Category 1 PAMS transmitters above the maximum DBA flood level within containment. This change is made to ensure that the 18 Category 1 PAMS transmitters are available following the DBA flood. [

]

The applicant also proposed to change the post-accident operation period for instruments that monitor Category 2 parameters. These changes would not affect Type A, B, or C primary post-accident parameters. These proposed design changes reduce the post-accident operation period from 4 months to 2 weeks. The Category 2 PAMS transmitter instruments are not required long term following a DBA, so their post-accident operation period can be changed from 4 months to 2 weeks. As Category 2 parameters, these parameters are not considered to be primary post-accident parameters for Type A, B, or C and, therefore, are not required to be qualified long-term following a DBA.

In the markup for AP1000 DCD, Tier 1, Table 2.2.2-1, the applicant changed the qualification for harsh environment from “Yes” to “No” for the seven containment pressure sensors. This change is not justified because the seven containment pressure sensors are still inside containment, even though the seven transmitters are moved outside containment. The applicant also failed to identify the 18 Category 1 PAMS transmitters that are proposed to be relocated above the maximum DBA flood level. As a result, the staff issued RAI-DCP-CN64-ICE-01 requesting the applicant to justify the changes in the environmental qualification of the seven containment pressure sensors. Also, the applicant was requested to identify which Category 1 PAMS transmitters are relocated.

In the RAI response, the applicant restored a harsh environment qualification requirement for the seven containment pressure sensors that are still located inside containment. In the response, the applicant also identified all 18 Category 1 PAMS transmitters that are relocated above the maximum DBA flood level. After reviewing all proposed design changes and information provided in the RAI response, the staff finds that the design changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

These proposed changes also affected the TS Bases for Engineered Safety Feature Actuation System Instrumentation, B 3.3.2. The staff found this change acceptable since it provides details of the system design described in the changes to Section 6.2.2 of the DCD.

23.J.4 Conclusion

After reviewing the proposed design changes, markups to DCD Tier 1 and Tier 2 and RAI responses, the staff finds that the proposed design changes meet the post-DBA monitoring requirements for the transmitters. Therefore, the staff concludes that the proposed design changes are acceptable.

23.K Changes to Startup Feedwater System and Chemical and Volume Control System Isolation Logic

23.K.1 Description of Proposed Changes

In letters dated May 10, 2010, and July 29, 2010, the applicant proposed design changes to add an AND logic to the protection and safety monitoring system (PMS) Functional Diagram Sheets 6 and 10 of Figure 7.2-1 in the AP1000 DCD to isolate the startup feedwater system (SFW) and close the CVS isolation valves earlier in the steam generator tube rupture (SGTR) transient sequence in order to maintain the margin to the SG overfill. The SG narrow range level high coincident with reactor trip limiting setpoint is proposed to be changed to 85 percent and the actuation signal added to Table 7.3-1, Table 15.0-6, and TS Table 3.3.2-1 and its Bases. The newly added AND logic is to combine SGS narrow range level high with P-4 reactor trip.

23.K.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes of SFW and CVS isolation on SGS narrow range level high coincident with P-4 reactor trip is documented in Chapter 7 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.3, “Engineered Safety Features Systems.” Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified by NUREG-0800 Section 7.3 (as applicable).

23.K.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the engineered safety features in Section 7.3 of AP1000 DCD Tier 2. The staff reviewed the applicant’s proposed design changes to the SFW and CVS isolation logic by using the review procedures described in NUREG-0800 Section 7.3 and requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A.

The containment pressurization backpressure was not considered in the original transient analysis for the DBA of SGTR. However, the increase of the containment pressure resulting from heat transfer to containment via the passive residual heat removal (PRHR) heat exchanger (HX) impacts the boiling temperature of the in-containment refueling water storage tank (IRWST) water and results in a longer duration of SGTR breakflow. The analysis also showed that the SG overfill occurs at 12,739 seconds during a design basis SGTR accident with no operator actions modeled. These design changes have been proposed in order to maintain SG overfill margin without reliance on operator actions.

For the specific DCD changes, the applicant mainly added an AND logic to the PMS software logic (Sheets 6 and 10 of Figure 7.2-1), which combines the existing SG narrow range level high signal from either of the two SGs and P-4 reactor trip signal to isolate SFW and close CVS isolation valves. The narrow range level high signal for each SG results from a coincidence logic of two out of four divisions with bypass functionality. This change meets the reliability and testability requirements in GDC 21, “Protection System Reliability and Testability.”

Sections 7.3.1.2.13 and 7.3.1.2.15 in AP1000 DCD Tier 2 are also revised to include the new isolation signal for the SFW and CVS systems. The SG narrow range level high limiting setpoint at 85 percent and the actuation signal are added to Table 7.3-1, Table 15.0-6 and TS Table 3.3.2-1 and its bases. The staff finds that these design changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made changes to the DCD text which are consistent with the proposed changes.

The AP1000 design provides automatic protection actions to mitigate the consequences of a SGTR event. The automatic actions include reactor trip, actuation of the PRHR HX, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of CVS and SFW on high SG narrow range water level. The proposed change to add the SG narrow level high coincident with P-4 reactor trip would isolate the CVC and SFW at a relatively lower SG level and is, therefore, conservative during the SGTR. This added trip isolation function would conservatively maintain the SG overfill margin without relying on operator actions.

The staff has reviewed the proposed change to add an AND logic to the PMS software that combines the existing SG narrow range level high alarm and P-4 reactor trip to isolate SFW and CVS. Based on its review, the staff finds that the results of the analysis continue to show that the SG will not overfill with water, the maximum RCS pressure will not exceed 110 percent of design pressure, and the minimum DNBR will remain greater than the safety DNBR limit. Therefore, the staff concludes that this proposed change is acceptable.

The proposed design changes also include the addition of isolation functions to the engineered safety feature actuation system (ESFAS). If a narrow range high SG level signal is received, coincident with a P-4 reactor trip, signals are sent to isolate the CVS and SFW. In the case of CVS, this prevents additional make-up to the RCS in case of a SGTR casualty. The SFW isolation prevents an overfill condition and, therefore, an over cooling condition, if the reactor is tripped and the SFW system is not isolated. These changes affect the TS for ESFAS instrumentation, Table 3.3.2-1, as well as the discussion in bases Sections B 3.3.2 and B 3.4.17. The staff finds these changes acceptable since they conform to the guidance provided in the Standard Technical Specifications (STS) and add appropriate isolation for protection in these events, as discussed in the changes to Section 15.6.3 of the DCD.

23.K.4 Conclusion

After reviewing the proposed design changes and markups to DCD Tier 2 sections and tables, the staff finds that the proposed changes meet the applicable requirements and guidance for the safety-related functions. Therefore, the staff concludes that these design changes are acceptable.

Based on the above evaluation, the staff concludes that the proposed design changes are acceptable because the SGTR analysis continues to show margin to SG overfill, and meet the pressure and core safety DNBR limits, even though they are not required to be met for a SGTR event.

23.L Changes to Passive Core Cooling System Injection Lines

23.L.1 Introduction

In letters dated May 25, 2010; August 2, 2010; and August 20, 2010, the applicant proposed design changes to the passive core cooling system (PXS) injection lines to address gas intrusion, which included the addition of manual vent valves, pipe stubs, manual drain valves, instrumentation, and re-routing accumulator discharge line connections to the direct vessel injection (DVI) lines. These manual vent valves are located in containment rooms that are constructed to permit entry during full power conditions. Analyses have shown that with enough noncondensible gas accumulation, IRWST injection through the affected flow path could be delayed. Therefore, excessive amounts of noncondensible gas accumulation in the high point vents of the IRWST injection lines may potentially impact the passive injection of IRWST borated water into the reactor vessel (RV). However, the presence of a small amount of noncondensible gases does not imply that the IRWST injection capability is immediately inoperable but rather that gases that are accumulating need to be vented. The venting of these gases requires containment entry to manually operate the vent valves. Since gas accumulation is a slow process, plant operators have sufficient time to vent the noncondensible gases upon receiving an alarm. To incorporate the proposed change, the system modification would include revisions to Tier 1 Figure 2.2.3-1, Tier 2 Figures 5.1-5 and 6.3-1, and Tier 2 Tables 3.2-3, 3.9-17, 3.11-1 and 3I.6-3 to identify the new components. In addition, the proposed change includes three new DCD Sections 6.3.6.3, 6.3.6.3.1, and 6.3.6.3.2, which provide a description of the plan to mitigate gas intrusion and accumulation by means of periodic system surveillance and venting procedures, review of pipe layout and routing drawings to identify high-point vent and low-point drain locations, and assessment of system design features. Also, to ensure proper operational implementation of this design change since gas accumulation has the potential to impact safety-related systems, IRWST required actions and surveillance requirements are added to DCD Section 16.1, "Technical Specifications." For controls of IRWST operations, the affected TS sections are TS 3.5.6, TS 3.5.7, and TS 3.5.8 for during Modes 1 through 4, Mode 5, and Mode 6, respectively.

23.L.2 Regulatory Basis

GDC 27, "Combined Reactivity Control Systems Capability" in 10 CFR Part 50, Appendix A, requires that the emergency core cooling system (ECCS) be designed to provide the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

GDC 27 is applicable because upon actuation the ECCS provides rapid injection of borated water to ensure reactor shutdown. Injection of borated water provides negative reactivity to reduce reactor power to residual levels and ensures sufficient cooling flow to the core. Meeting the requirements of GDC 27 for the ECCS augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident conditions, can be safely shut down and will be maintained in a coolable geometry. Noncondensible gas accumulation has the potential to delay injection of borated water. Such a delay would impact the moderating and heat removal capabilities, thus providing an adverse challenge to the primary fission product barrier and maintenance of coolable core geometry.

GDC 35, "Emergency Core Cooling," requires, among other things, that the ECCS be designed to provide an abundance of core cooling to transfer heat from the core at a rate such that fuel and clad physical damage that could interfere with continued effective core cooling is prevented.

GDC 35 is applicable because following a breach in the reactor coolant pressure boundary; reactor coolant is lost at a rate determined by several factors, including break size and RCS pressure. The ECCSs are relied upon to inject adequate cooling water into the RCS during a LOCA and to circulate the water through the core to provide for core cooling. The ECCS must inject cooling water at a rate sufficient to ensure that the calculated changes in core geometry will be such that the core remains amenable to cooling, and that the calculated cladding oxidation and hydrogen generation meet the specified performance criteria. Based on analysis, noncondensible gas accumulation has the potential to delay injection of borated water; such a delay may adversely affect fuel and cladding physical configuration with potential to challenge the coolability of the core geometry.

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the IRWST design and operating information described in DCD Section 6.3. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified in NUREG-0800 Chapter 16.

23.L.3 Evaluation

Noncondensible gas accumulation effects on the performance of safety related "active" emergency core cooling, residual heat removal, and containment spray systems have been documented based on operating plant experience and system analysis. Some of these events are identified in NRC generic letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems," which illustrates the need for license holders to review their safety systems to resolve this issue. The focus of this GL is centered on four principal concerns: (1) licensing basis, (2) design, (3) testing, and (4) corrective actions. Also, the GL provides an assortment of pertinent generic communications documents related to this subject, which was helpful in the staff's evaluation. Although the GL refers to active safety systems focusing on gas accumulation effects on pump performance, the applicant expanded its existing design activity to include AP1000 passive safety systems to "integrate the draft Interim Staff Guidance (ISG) document ISG-019 regarding gas intrusion assessment guidance into the design process, helping to confirm that the potential issues identified in" the GL have been addressed in the AP1000 design. However, this safety evaluation is related to proposed design changes that primarily address gas intrusion and accumulation in the PXS.

The proposed changes include the addition of the following components:

- 8 manual maintenance vent valves in 6 PXS passive injection and recirculation line piping high point locations;

- 4 pipe stubs with maintenance vents and associated valves, line routing to tee into core makeup tank (CMT) vent line routing to the reactor coolant drain tank (RCDT);

- Remote pipe stub gas void indications at the outlets of each of the IRWST passive injection squib valves;

- Re-routing the 2 accumulator discharge line connections to both DVI line vertical cold trap (riser) pipes to tee in physically above (in elevation) and downstream of the associated IRWST connection into the DVI riser pipes (instead of locating the accumulator connection upstream of and below the IRWST tee);

- 20 manual maintenance drain valves in 14 PXS passive injection and recirculation piping locations, 5 normal RNS piping locations, and 1 RCS piping location. (Note: The RNS and RCS drains are unrelated to gas intrusion effects but part of design finalization to consolidate required PXS line changes and piping analyses.)

Issues identified during staff review were addressed in three RAIs, identified as RAI-DCP-CN66-SRSB-01 through RAI-DCP-CN66-SRSB-03, with the primary issues summarized below:

1. RAI-DCP-CN66-SRSB-01
   - Explain the exclusion of vent valves to Train B of the containment recirculation.

2. RAI-DCP-CN66-SRSB-02
   - Provide the bases for selecting 5.7 Liter (L) (0.2 cubic feet ($ft^{3)}$) as the noncondensible gas volume limit in the TS surveillance requirements (SR).
   - Describe the volume measuring method of the noncondensible gas.
   - Describe how the 5.7 L (0.2 ft^3) volume of noncondensible gas is accounted for in the safety analyses for LOCA and post-LOCA long term cooling.

3. RAI-DCP-CN66-SRSB-03
   - Provide justification for excluding the new instrumentation and valve components in ITAAC.
   - Describe the calibration frequency of resistance temperature detector (RTD) switches used to measure the volume.
   - Discuss whether the calibration frequency is controlled by TS.

In its August 20, 2010 response to RAI-DCP-CN66-SRSB-01, the applicant stated that containment recirculation Train A and Train B paths have different configurations to avoid other components and piping lines. Whereas, Train A was routed with two high points to circumvent interference with other plant components, Train B layout constraints were less complicated; hence, there are no local high points that could accumulate gas. After further review, the staff finds the RAI response acceptable.

With regard to RAI-DCP-CN66-SRSB-02, the applicant responded that originally the 5.7 L (0.2 ft^3) noncondensible gas volume limit was selected for the CMT inlet high point pipe TS SR 3.5.2.4 because it was slightly larger than the internal volume between the sensors location on the pipe stubs and the first normally closed manual vent valve that was previously evaluated and approved. Since the gas intrusion and accumulation is a slow process and the alarm occurs before voiding is extended into the line, the relocation allows sufficient time for the operators to vent the line before gas buildup could adversely affect the passive safety operability performance. Therefore, initially the PXS volume limit of 5.7 L (0.2 ft^3) was selected to be consistent with the CMT SR requirements since the IRWST injection line high points, pipe stubs, and sensor configuration is similar to that of the CMT. However, based on the applicant's response, the staff concludes that the sensor location not the volume limit is relevant to proper performance of venting the line because the sensor is configured as a level switch, as discussed below. Therefore, the applicant has proposed to remove the reference to the volume limit from TS SRs 3.5.2.4, 3.5.4.3, and 3.5.6.3 and replace it with "has not caused the high-point-water level to drop below the sensor." Also, the corresponding TS Bases would be revised to appropriately reflect the removal of the volume limit. The staff finds this proposed change acceptable because it refers to the correct function of the sensor.

In addition, the applicant stated that the volume is not directly measured but the change of state from liquid to gas provides the mechanism that controls the sensor output. The sensors are thermal dispersion sensors consisting of one heated RTD and one non-heated RTD configured such that the RTDs function as a thermal dispersion level switch where the temperature difference is based on the conductivity of the medium in contact with the two elements. When the RTDs are exposed to gas, the change in the differential temperature of the elements causes the output switch to actuate providing an alarm to the operator. This sensor configuration has been used in other plant applications with reliable results. Therefore, the staff finds this method to be acceptable.

With respect to safety analyses for LOCA and post-LOCA long term cooling, the applicant stated that the proposed volume is not considered in any safety analyses because the pipe stub and sensor configuration allows for sufficient time for venting before the actual injection path begins to void. The staff finds this acceptable because the alarms trip with less than half the volume of gas that could affect the injection flow performance, the rate of gas accumulation is expected to be sufficiently slow, and there are no credible postulated gas intrusion mechanisms for these locations.

In its August 20, 2010, response to RAI-DCP-CN66-SRSB-03, the applicant stated that the proposed instrumentation switches provide pre-event operability confirmation and are not used for reactor trip, safeguards actuation, or post-accident monitoring functions. As such, these switches are not required to be part of ITAAC or TS. Also, since the switches are nonsafety related process instrumentation, there is limited periodic calibration or functional checking requirements. The staff considers acceptable the response that the process switches do not meet the requirements to include them in ITAAC and TS.

The proposed changes also included the addition of two new actions and one SR to TS 3.5.6, TS 3.5.7, and TS 3.5.8 related to one or two IRWST injection line(s) “inoperable due to presence of noncondensible gases” and surveillance of the new sensors indication. Also, the appropriate TS Bases would be revised to reflect this change. The staff reviewed the new actions, SR, and bases, which provide for conditions, required actions, completion times, and SRs. The staff found that these parameters conform to guidance provided in the STS and reflect the IRWST injection line gas intrusion and accumulation process analysis described in DCD Section 6.3. Therefore, the staff finds the TS change, as modified in the response to RAI-DCP-CN66-SRSB-03 noted above, to be acceptable.

In addition, the proposed change would provide for system surveillance and venting procedures, as described in DCD Chapter 13, to include inspection of the passive safety system location equipped with manual vent valves to eliminate any identified gas accumulation. Procedural inclusion of vent component inspection is acceptable to the staff because timely inspections could reduce intrusion and accumulation of gases in the injection lines.

In accordance with NUREG-0800 Section 6.3, the staff’s evaluation of these proposed design changes must ensure that compliance with GDC 27 and GDC 35 is satisfied. As stated in NUREG-0800, meeting the requirements of:

> …GDC 27 for the ECCS augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident

> conditions, can be safely shut down and will be maintained in a coolable geometry.
>
> …GDC 35 ensures that the ECCS, assuming a single failure, can provide core cooling under accident conditions sufficient to maintain the core in a coolable geometry and to minimize the reaction of water with the fuel cladding.

The staff evaluation of the proposed change concludes that the reliability and performance of PXS is improved. The requirements of GDC 27 and GDC 35 are satisfied based on the following:

- TS requirements that ensure that the PXS is operated in safe condition;
- Adequate procedural inspections to identify and eliminate gas intrusion and accumulation; and
- Sufficient monitoring and alarming features in the control room to ensure timely venting.

23.L.4 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes to provide venting of noncondensible gases in the injection lines enhance the reliability and performance of the PXS to ensure that the core, under postulated accident conditions, can be safely shut down, maintain a coolable geometry, and minimize the reaction of water with the fuel cladding, thus satisfying the requirements of GDC 27 and GDC 35. Therefore, the proposed design changes are acceptable.

23.M Changes to Squib Valve Actuation Time

23.M.1 Description of Proposed Changes

In letters dated April 28, 2010, July 29, 2010, and August 12, 2010, the applicant proposed design changes to the PMS and diverse actuation system (DAS) controls for PXS IRWST injection squib valves actuation by incorporating a 5-second time delay in PXS actuation control for the automatic (and manual) actuation circuitry between the firing of the first and second valve for each pair of squib valves in the same process line. These proposed design changes address the unacceptable design stress on DVI line piping during simultaneous actuation of squib valves in the parallel configuration. The applicant also made similar changes to PMS controls for PXS containment sump recirculation squib valves in parallel path of the same process line. These proposed design changes revise functional diagrams in DCD Figure 7.2-1 by adding Note 5 in Sheet 16, “In-containment Refueling Water Storage Tank Actuation,” and Note 6 in Sheet 20, “Diverse Actuation System Logic, Manual Actuation,” which state that, for redundant components in a parallel configuration, the components use different time delays to prevent simultaneous actuation.

23.M.2 Regulatory Basis

GDC 35 in 10 CFR Part 50, Appendix A, requires that an ECCS be provided to transfer heat from the reactor core following any LOCA at a rate such that: (1) fuel and clad damage that could interfere with continued effective core cooling is prevented; and (2) clad metal-water reaction is limited to negligible amounts. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specifies that the calculated ECCS cooling performance be shown to comply with the acceptance criteria specified in 10 CFR 50.46(b).

The regulatory basis for evaluating the changes in squib valve actuation time is documented in Chapters 6, 7, and 15 of NUREG-1793.

Reviews of the proposed changes are also based on meeting the relevant requirements of 10 CFR 50.55a(h) and 10 CFR 52.47. The changes should also conform to the guidelines of Institute of Electrical and Electronic Engineers (IEEE) Standard 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by RG 1.152, "Criteria for Use of Computer in Safety Systems of Nuclear Power Generating Stations," and the staff requirements memorandum (SRM) on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.3, "Engineered Safety Features Systems," and Section 7.8, "Diverse Instrumentation and Control Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Sections 7.3 and 7.8 (as applicable).

23.M.3 Evaluation

The proposed changes add notes in the PMS and DAS functional diagrams in DCD Figure 7.2-1 to state that the redundant components in a parallel configuration use different time delays to prevent simultaneous actuation from adversely impacting accident timing in the safety analyses. For the automatic depressurization system (ADS) stage-4 squib valves, DCD Table 15.6.5-10 specifies the actuation time delay of 60 seconds between the ADS-4A and ADS-4B squib valves. The proposed change was only required for the IRWST injection and containment recirculation sump squib valves since the ADS stage-4 valves already have a 60-second time delay difference in their automatic actuation circuitry.

In its August 12, 2010, response to RAI-DCP-CN08-SRSB-01, the applicant stated that its proposed 5-second time delay for the actuation of redundant squib valves in each IRWST injection path and containment recirculation flow path, respectively, was included not to correct the system safety performance, but rather to protect the integrity of components, piping and structural and mechanical modules, in relationship to the specific squib valves actuation characteristics. The applicant performed an evaluation of LOCA safety analyses with the 5-second delay and found that the impact is not significant. The AP1000 design has the squib valves actuate a relatively long time before they are needed, on the order of several hundreds of seconds, so that the very short squib valve actuation delay of 5 seconds or reasonably longer time for the second squib valve in each path is not significant to the IRWST injection and/or containment recirculation performance of the plant events. Therefore, the safety analyses have not been revised to implement this 5-second time delay. The staff agrees with this explanation. The existing safety analyses in DCD Section 15.6.5 for the large-break LOCA, small-break LOCA, and long-term cooling remain valid and in compliance with GDC 35 and 10 CFR 50.46. Therefore, this proposed change is acceptable.

As documented in NUREG-1793, the staff reviewed and approved Functional Diagram Sheet 16 of Figure 7.2-1 for PXS IRWST injection and containment recirculation isolation squib valves actuation and Functional Diagram Sheet 20 of Figure 7.2-1 for DAS manual actuations for IRWST injection squib valves. The staff reviewed the applicant's proposed design changes to the two functional diagrams using the review procedures described in NUREG-0800 Sections 7.3 and 7.8.

10 CFR 52.47 requires that an application include a sufficient description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification; therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Hence, the staff issued RAI-DCP-CN08-ICE-02, requesting the applicant provide clarification for the time delay inconsistency between the design change description and marked-up diagrams and also justify the need for a 5 second time delay between the firing of the two IRWST injection squib valves. In the RAI response, the applicant explained that the 5-second time delay is required to prevent a negative structural impact on the supporting bracket from induced vibrations if both squib valves are fired simultaneously. The firing of one of the commonly mounted explosive valves followed by a five second time delay will have no negative impact on operation. The design change provides an improved protection of the safety function of the IRWST injection and containment recirculation by avoiding potential adverse consequences related to the simultaneous firing of the two squib valves physically housed on the same structural frame module. The staff finds that the design changes and response to the RAI are acceptable. Therefore, RAI-DCP-CN08-ICE-02 is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made changes to the DCD text which are consistent with the proposed changes.

23.M.4 Conclusion

The staff concludes that the proposed changes of adding notes to the PMS and DAS function diagrams in DCD Figure 7.2-1 are acceptable because they have negligible effect on the LOCA safety analysis, and continue to comply with GDC 35 and 10 CFR 50.46.

In addition, the staff concludes that the applicant’s proposed design changes are acceptable because they have no negative impact on operation, and provide improved protection of the integrity of components, piping and structural and mechanical modules.

23.N Changes Related to Anticipatory Reactor Trip in the Event of an Inadvertent Passive Residual Heat Removal Actuation

23.N.1 Description of Proposed Changes

In a letter dated May 10, 2010, and in two letters dated July 29, 2010, the applicant proposed design changes related to anticipatory reactor trip in the event of an inadvertent PRHR actuation. The applicant proposed design changes to the PMS to include one additional reactor trip to mitigate an inadvertent PRHR event caused by the opening of either of the two PRHR HX discharge valves when the reactor is in power operation. In the PMS logic control system, a reactor trip that is to be added will be generated when either of the two PRHR HX discharge valves comes off its fully shut seat while the reactor is at power. Another proposed design change is to adjust the frequency of the inservice test (IST) for the two PRHR HX discharge valves from the current once every quarter to every cold shutdown.

In order to mitigate an inadvertent PRHR actuation event, the applicant proposed changes to add a PRHR actuation reactor trip function, which is based on the PRHR HX control valve (CV) indication, to DCD Chapter 7.2, "Reactor Trip." Related to this design change, a sentence is added to DCD Section 6.3.7.6.1, which states that for the PRHR HX discharge valves, valve position indication is used to initiate a reactor trip upon opening of these valves while the reactor is at power. Also, DCD Section 15.1.6, "Inadvertent Operation of the PRHR Heat Exchanger," is revised to state that to prevent the reactivity increase (as a result of inadvertent actuation of the PRHR HX event) from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off of its fully shut seat. DCD Section 15.1.6.2 is revised to state that since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis. In DCD Table 15.0-4a, the reactor trip function on the PRHR discharge valve not closed with a time delay of 1.25 seconds is added, and in DCD Table 15.0-6, this reactor trip function is listed for the inadvertent operation of the PRHR event.

23.N.2 Regulatory Basis

The regulatory basis for evaluating the proposed changes to the AP1000 PMS reactor trip system is documented in Chapters 7, 15, and 16 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of regulations and guidelines in 10 CFR 50.55a(h); 10 CFR 52.47; and GDC 20 and GDC 21 of 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.2, "Reactor Trip Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Section 7.2 (as applicable).

GDC 10 in 10 CFR Part 50, Appendix A, specifies that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. An inadvertent operation of the PRHR heat exchanger is an AOO and, therefore, must comply with GDC 10.

23.N.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the reactor trip system as specified in Section 7.2 of AP1000 DCD, Revision 15. The staff reviewed the applicant's design changes to the reactor trip system using the review procedures described in NUREG-0800 Section 7.2.

If either of the two PRHR HX discharge valves inadvertently comes off its normally fully closed seat while the reactor is at power, it allows a slug of cold water to be introduced into the reactor core through the PRHR HX to cause a marked increase in reactor power, which may exceed the fuel design limits. Therefore, a reactor trip is needed to mitigate this inadvertent PRHR event introduced by the opening of either of the two PRHR HX discharge valves, and prevent the fuel design limits from being exceeded. However, in the event of such an inadvertent PRHR event, the current reactor trip design included in the AP1000 DCD, Revision 17 will not be able to function fast enough to mitigate this inadvertent PRHR event. Therefore, the proposed design changes result in an improved reactor protection system to mitigate the inadvertent PRHR event. In addition, the 2-out-of-4 control logic with bypass capability is also used for this reactor trip condition in this DCP. The staff finds that all those changes meet applicable requirements in

10 CFR 50.55a(h), GDC 20, and GDC 21. The applicant made necessary changes to DCD Tier 1 Section 2.5.2, Tier 2 Section 7.2, and other related tables and figures. The staff finds that the proposed changes to reactor trip logic for an inadvertent PRHR actuation are acceptable because they satisfy the requirements of 10 CFR 50.55a(h), GDC 20, and GDC 21. In a subsequent revision to the AP1000 DCD, the applicant made changes to the DCD text, which are consistent with the proposed changes.

An inadvertent operation of PRHR HX causes an injection of relatively cold water into the RCS, resulting in a reactivity insertion due to a negative moderator temperature coefficient. Currently, several reactor trip functions are available to mitigate the event, including the overpower ΔT and overtemperature ΔT trip functions, to prevent a power increase, which could lead to a DNBR less than the safety analysis DNBR limit. The safety analysis of the limiting inadvertent operation of PRHR HX, presented in DCD Section 15.1.6, shows that, without taking credit for a reactor trip function, the nuclear power rises to about 120 percent temporarily, then drops and reaches a new equilibrium condition at about 108 percent of the nominal value. The RCS pressure and minimum DNBR are within the respective limits.

The proposed design change to add a PRHR actuation reactor trip function will enhance the reactor protection to trip the reactor upon event initiation. In its July 29, 2010, response to RAI-DCP-CN60-SRSB-01, the applicant stated that it has performed an evaluation of the operation of the PRHR HX transient assuming a conservative 1.25 second reactor trip response time, which covers the time to sense the opening of a PRHR valve through initial insertion of control rods. A confirmatory analysis using LOFTRAN shows that the minimum time between the opening of the PRHR valves and any colder water reaching the reactor core inlet is at least 2 seconds. Since the control rods are already inserting into the core before any colder water reaches the core inlet, there will not be an adverse power increase for this transient. Since an inadvertent operation of the PRHR HX event can only be caused by an inadvertent opening of the PRHR HX discharge valves (either by operator error, false actuation signal, or malfunction of a discharge valve), and a reactor trip will be initiated as soon as a PRHR discharge valve comes off the closed seat position, the consequence is mitigated as soon as the inadvertent PRHR HX operation event occurs. The proposed design changes continue to comply with SAFDL and no additional safety analysis is needed.

Each of the two PRHR HX discharge valves has one set of existing Class 1E magnetic valve position indicators (VPIs), one for the open position indication and one for the close position indication. The design change will use the existing close position indicator and add three more Class 1E magnetic VPIs to give a total of four closed signals per valve. This configuration is necessary for the “two out of four” logic required for a reactor trip signal. The proposed change in DCD Section 6.3.7.6.1 merely indicates that the PRHR HX discharge valves’ position indication is used for the reactor trip function, and is, therefore, acceptable.

The applicant’s proposed design changes add a reactor trip signal generated from the opening of valves in the PRHR system. This trip would minimize the effect of the anticipated reactivity excursion due to cold water addition by the PRHR system initiating flow. This change affected the TS for reactor trip system instrumentation, Table 3.3.1-1, as well as discussions in bases Sections B 3.1.1 and B 3.3.1. The additional information included in this change added appropriate descriptions of the trip function and purpose, and created a trip function in Table 3.3.1-1 that provides adequate protection against this event. The staff finds this change acceptable since it conforms to the format and trip functions described in the STS.

The applicant's proposed design changes also include changing the inservice testing full-stroke exercising requirement for the PRHR HX valves PXS-V108a and PXS-V108b from once a quarter to every cold shutdown. Valves PXS-V108a and PXS-V108b are Class 1, Category B air-operated valves that function as discharge valves for the PRHR HX. The inadvertent actuation of the PRHR HX causes an injection of relatively cold water into the RCS. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. To prevent this reactivity increase from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off its full shut seat. The staff considers changing the IST full-stroke exercising requirement for valves PXS-V108a and PXS-V108b from once a quarter to every cold shutdown to be acceptable based on the ASME Code (2004E) ISTC-3521(c), which states that if exercising is not practicable during operation at power, it may be limited to full-stroke testing during cold shutdown.

23.N.4 Conclusion

After reviewing the proposed changes and markups to associated DCD Tier 1 and Tier 2 sections, tables, figures, and TS, the staff finds that the proposed changes provide an improvement to the reactor protection system. The staff concludes that the proposed changes to add a PRHR actuation reactor trip function is acceptable because it will trip the reactor as soon as an inadvertent operation of PRHR HX event occurs, thereby ensuring that the SAFDL will not be exceeded. Therefore, the staff concludes that these proposed design changes are acceptable.

23.O Changes to Reactor and Turbine Trips Functional Logic of Diverse Actuation System

23.O.1 Description of Proposed Changes

In letters dated May 20, 2010, and July 29, 2010, the applicant proposed design changes to add a reactor trip and turbine trip to the functional logic of the DAS based on the 2-out-of-2 control logic of the PRHR high hot leg temperature sensor outputs.

23.O.2 Regulatory Basis

The regulatory basis for evaluating these proposed design changes of DAS PRHR high hot leg temperature logic is documented in Chapter 7 of NUREG-1793.

Reviews of the changes are also based on meeting the relevant requirements of regulations and guidelines in 10 CFR 50.55a(h) and 10 CFR 52.47. The changes must also conform to the requirements of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 7.8, "Diverse Instrumentation and Control Systems." Acceptability was based on conformance with the existing AP1000 licensing basis and criteria specified in NUREG-0800 Section 7.8 (as applicable).

23.O.3 Evaluation

As documented in NUREG-1793, the staff reviewed and approved the DAS in AP1000 DCD Tier 1 Section 2.5.1 and Tier 2 Section 7.7.1.11. The DAS system in the certified AP1000 DCD, Revision 15, uses the 2-out-of-2 control logic, which is based on the PRA evaluation. The staff reviewed the proposed design changes using the review procedures described in NUREG-0800 Section 7.8 and applicable regulations of GDC 13 and GDC 20 in 10 CFR Part 50, Appendix A.

According to the modeling in the PRA, the original DAS design in the AP1000 DCD should initiate a reactor trip and turbine trip for anticipated transients without scram (ATWS) sequences with the main feedwater available. Since the main feedwater is still available, the low SG water level signal will not be generated to initiate the reactor or turbine trip in the DAS. Therefore, the high hot leg temperature signal is needed in the DAS to trip the turbine or the reactor via the control rod motor-generator (MG) sets.

10 CFR 52.47 requires that an application include a sufficient description and analysis of the SSCs of the facility, with emphasis upon performance requirements, the bases, with technical justification; therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Hence, the staff issued RAI-DCP-CN63-ICE-01 requesting the applicant provide justification for adding another time delay for opening PRHR discharge valves, which is not explained in the DCP. The applicant's July 29, 2010, RAI response stated that the time delay has been provided in the functional design to support sequencing of the output field devices associated with a system-level control function. This delay time is added to be consistent with the use of timers in the existing functional design (e.g., low SG water level logic). The staff finds that the response to the RAI is acceptable.

After reviewing the proposed design changes and RAI responses, the staff finds that the proposed design changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.O.4 Conclusion

After reviewing the proposed design changes, including markups to DCD Tier 1 and Tier 2 and RAI responses, the staff finds that the proposed changes will meet the required DAS functions as modeled in the PRA and related regulatory criteria. Therefore, the staff concludes that these proposed design changes are acceptable.

23.P Changes to Steam Generator System Instrument Piping

23.P.1 Description of Proposed Changes

In a letter dated July 8, 2010, the applicant proposed design changes in the material for the SGS instrument piping. These design changes would modify Figure 10.3.2-1 of the AP1000 DCD to specify stainless steel piping for all AP1000 Quality Classes B and C instrument piping for the SGS. AP1000 Quality Classes B and C are designed and fabricated to ASME Code, Section III, Class 2 and 3, respectively. The main steam supply system (MSSS) includes the AP1000 SGS.

23.P.2 Regulatory Basis

The staff reviewed and evaluated the proposed design changes in accordance with the guidance in NUREG-0800 Section 10.3.6 to ensure that the ASME Code, Class 2 and 3 MSSS components use material specified in Sections II and III of the ASME Code, thereby meeting the requirements of GDC 1, “Quality Standards and Records,” and 10 CFR 50.55a.

23.P.3 Evaluation

Figure 10.3.2-1 modified all of the ASME Code, Section III, Class 2 and 3 SGS instrument lines, which typically are 2.5 cm (1 in) NPS and less, to be a corrosion resistant (stainless steel) material. Section 10.3.6.2 of the AP1000 DCD specifies that the material selection and fabrication for ASME Code, Section III, Class 2 and 3 are in accordance with the requirements of ASME Code, Section III, Class 2 and 3 components outlined in Sections 6.1.1.1 and 6.1.1.2 of the AP1000 DCD. Section 6.1.1.1 of the AP1000 DCD specifies that pressure-retaining materials meet the requirements of Articles NC-2000 and ND-2000 of the ASME Code, Section III for Class 2 and 3 components, respectively. The staff notes that Articles NC-2000 and ND-2000 for ASME Code Classes 2 and 3, respectively, specify that the material be in accordance with Section II of the ASME Code. Based on this information, the staff finds that the use of stainless steel material specified in ASME Code, Section II for ASME Code Classes 2 and 3 SG instrumentation piping, typically 1 NPS or less, is acceptable. The staff’s acceptance is based on the use of stainless steel for the instrumentation piping, which is more corrosion resistant than carbon steel, and that the material will be procured in accordance with ASME Code Section II and fabricated in accordance with ASME Code Section III. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.P.4 Conclusion

The staff concludes that the proposed use of stainless steel instrument piping for the SGS provides higher corrosion resistance than carbon steel, and that the material will be procured and fabricated in accordance with ASME Code Sections II and III as specified in the guidance of NUREG-0800 Section 10.3.6 and, therefore, meets the requirements of GDC 1 and 10 CFR 50.55a. The staff concludes that the proposed design changes are acceptable.

23.Q Changes to the Steel Containment Vessel Girder and Polar Crane Rail Clip

23.Q.1 Description of Proposed Changes

In letters dated April 26, 2010; August 26, 2010; and September 16, 2010, the applicant proposed design changes to the steel containment vessel (SCV) girder and polar crane rail clip. These design changes apply to the safety-related structure (containment vessel) and the heavy load handling system (heavy load lifting equipment and support) and include the following changes:

- Increase in the thickness of the girder top plate from 3.8 cm (1.5 in) to 4.45 cm (1.75 in)
- Change in the rail support from Gantrex Pad to a bolted clip design
- Extension of the SCV girder inward by 6.98 cm (2-3/4 in)

23.Q.2 Regulatory Basis

Sections 3.8.2 and 9.1.5 of NUREG-1793, address the function and acceptability of the SCV associated support structures for the overhead heavy load handling system in the AP1000 standard design. Acceptability of the proposed design change is evaluated to determine whether the design meets the relevant requirements specified in 10 CFR Part 50, Appendix A, GDC 2, which requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

23.Q.3 Evaluation

The SCV associated support structures are designed to provide the necessary support for the overhead heavy load handling (polar crane) system in the AP1000 standard design. The polar crane systems are safety-related and are used to handle heavy equipment, such as the integrated head package, SGs, and HXs. As stated in the DCD changes included in this design change, the design of the SCV associated support structures for the polar crane system support is consistent with the SSE design of equipment anchorages of seismic Category I equipment. The girder top plate and the new bolted clip design for the rail support are analyzed to show that the structure can withstand an SSE event.

During the staff's evaluation of the proposed design changes to the SCV associated support structures, it was determined that additional information was required from the applicant to complete the staff's evaluation. The staff generated RAI-DCP-CN-5-SEB-01 requesting: (1) an analysis including materials and design basis loading used to demonstrate that the increase of 0.64 cm (1/4 in) from 3.81 cm (1-1/2 in) to 4.45 cm (1-3/4 in) thick in top plate of the crane girder is adequate to support the design-basis loadings including a seismic load; (2) an analysis to demonstrate the new rail clip design, including clip spacing is sufficient to meet the design-basis load requirements including the seismic load demands; and (3) the basis for determining the extent of the inward extension of the CV girder top plate.

The applicant's response to RAI-DCP-CN-5-SEB-01 provided sufficient information on the design analysis for the proposed design changes to support the staff's evaluation. In addition to evaluating the applicant's proposed design changes and RAI responses, the staff performed two audits.

With respect to the SCV polar crane girder top plate design, the applicant presented a design analysis, which is documented in the technical report APP-MV50-S2C-020, Khanh Do, "Polar Crane Girder Top Plate Analysis." The stress analysis considers a total of 192 load combinations. Using ANSYS 11.0, a commercially available general purpose code, a 3D finite element model was constructed with the boundary conditions fixed at Elevation (El.) 30.5 m (100 ft) (i.e., at the grade level) for the purpose of determining the maximum stress intensities at the top plate. It was found that the maximum stress intensity [

] is well within the allowable stress intensity of 90 kips per square inch (ksi) (620.5 MegaPascal (MPa)) according to ASME NE-3221 in ASME Code Section III, Division 1, Subsection NE, Class MC, 2001 Edition. Accordingly, a 4.45 cm (1.75 in) thick plate is shown to be adequate to support the design-basis loadings, because the maximum stress intensity generated in the top plate is less than the allowable stress intensity the steel plate can offer. The staff reviewed the methodology of the analysis including model construction, boundary and loading conditions, material properties and applicable code; and determined that the design analysis is acceptable, and the use of a 4.45 cm (1.75 in) thick top plate is in compliance with the ASME Code requirements.

With respect to the new rail clip design with the bolted clips, the applicant presented an analysis in PR-08-5020/70587483, Roger Johnson, "Polar Crane Mechanical Calculations," Revision 1, March 5, 2010, to show that the proposed new design is capable of resisting a horizontal wheel load [ ] due to seismic conditions. With the help of MathCad Version 13 and GTSTRUDL 29.1 seismic analysis, it was found that the rated load [ ] governs the main hoist system, whose design meets NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" requirements. The new clip design is given in Westinghouse drawing APP-MH01-V2-021 and Westinghouse drawing APP-MH01-V6-041 and is based on the adopted 76.20-cm (30-in) wheel diameter. [ ] With the use of the clip plate [ ] with three slotted holes, the maximum bending stress [ ] is less than the 81 ksi allowable specified by the code. The clip plate is bolted [ ] to the filler plate welded to the girder top plate [ ] and the induced shear [ ] will be within the allowable shear stress of 148.9 MPa (21.6 ksi) based on the American Welding Society (AWS) code (the ASME NOG-1 code allowable is 50 percent higher, i.e., 223.4MPa (32.4 ksi)). The staff reviewed the detailed calculations and found the new design as given in drawings APP-MH01-V2-021 and APP-MH01-V6-041 to be in compliance with the AWS and ASME Codes; thus, it is acceptable to the staff.

The third item of design changes is concerned with the inward extension of 6.98 cm (2-3/4 in) of the CV girder top plate. The reasoning, according to the submittal, is that the tolerance requirement of the polar crane rail is much tighter than the SCV girder. The staff requested the applicant provide the basis to justify that the increase of 6.98 cm (2.75 in) inward of the SCV girder top plate is quantitatively adequate to meet the tolerance requirement. In its response, the applicant provided ASME Code, ASME-NOG-01-1998 (NOG-1) as the basis in the polar crane design documented in Westinghouse calculation APP-GW-GEE-513, DCP3513, "Steel Containment Vessel Girder and Polar Crane Rail Clip Design." The code NOG-1 specifies that the maximum and minimum span between the rails must be no greater or less than the nominal span of the rails +/- 0.76 cm (3/8 in). This tolerance is tighter than the girder, based on construction experts, who specify the tolerance for the radius of the SCV girder to be +/- 4.95 cm (1.95 in). Because of this difference in tolerance requirements, the girder, including top plate, web and bottom plate, must be extended radially inward toward the center of the CV. Figure 5 in Westinghouse calculation APP-GW-GEE-513 provides the new design showing the tolerance stack-up and the required extension of the top plate. Figure 1, Note B and Figure 2 show the associated design changes. The staff reviewed the drawings and the new design, and found the design changes are in compliance with the ASME Code NOG-1, and thus, are acceptable.

The proposed design changes, including DCD markups and RAI responses, are acceptable. RAI-DCP-CN-5-SEB-01 is resolved.

23.Q.4 Conclusion

The staff concludes that the proposed design changes are acceptable because the proposed changes will not adversely affect safety-related SSCs and the capability of the polar crane systems to perform their intended functions of heavy load lifting and transportation. The proposed design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable.

23.R Changes to the Reactor Vessel Support System

23.R.1 Description of Proposed Changes

In letters dated May 10, 2010, and September 9, 2010, the applicant proposed design changes related to the structural support system for the RV, which is a safety-related structure. These design changes involve the following modifications of the RV support system:

- Eliminate reliance on the CA04 structural module as part of the reactor pressure vessel (RPV) support system.
- Provide support boxes as RPV supports
- Change the anchor supports from a “bolted into embedded plates” configuration to a “anchored directly to primary shield wall concrete base via steel embedment plates” configuration.
- Increase the length of the RPV support boxes or legs.
- Install wear plates for the RPV bearing and tribological performance (i.e., to support RV and reduce frictional wear due to thermal expansion).

23.R.2 Regulatory Basis

Sections 3.8.3 and 5.4.10 of NUREG-1793, address the function and acceptability of the RPV support structural box in the AP1000 standard design. Acceptability of the design change was evaluated for its conformance with the existing AP1000 licensing basis.

23.R.3 Evaluation

In the AP1000 standard design, the RPV structural support system is designed to provide the necessary support for the RPV. The original anchorage design was bolting into embedded plates of the CA04 structural module. Design finalization analyses determined that the resulting stresses exceed the allowable stresses. As a result, the applicant proposed design changes in which the overstressed CA04 module is not used to support the RPV. Instead, the applicant has proposed a revised design where four support boxes are used to support the RPV. There are four support “boxes” or “legs” located at the bottom of the RPV’s cold leg nozzles. The support boxes are anchored directly to the primary shield wall concrete base via steel embedment plates. As stated in the proposed design changes, the design of the RPV associated support structures is consistent with the SSE design of seismic Category I equipment. The RPV support boxes, including the new anchorage design, are analyzed to show that they can withstand an SSE event.

During the staff’s evaluation of the proposed design changes to the RPV associated support structures, it was determined that additional information was required from the applicant. The staff requested additional information related to: (1) a stress analysis of the proposed design, including materials and design basis loading used to demonstrate that the maximum stresses incurred at the critical sections of the RPV support box structure are lower than those of the previous design, and that the new design is in compliance with the adopted code requirements; (2) specifications of the wear plates, including material, lubricant used, geometric dimensions in size and shape, and performance requirements; and (3) an assurance of satisfactory tribological performance that uneven settlement on the plate's top surface will not deter lateral movement of the RV due to thermal expansion, and that the wear plate can endure the frictional wearing due to cyclic motion of the heavy RPV over its entire design life without loss of its intended design function.

In a letter dated September 9, 2010, the applicant provided sufficient information on the design analysis to support the staff's evaluation. The supporting information also included documentation on the material specifications of the self-lubricating bearings used for the wear plates.

With respect to the RPV support box design, the applicant stated that the design is in compliance with the design specification depicted in APP-SS30-Z0-001, "Design Specification," Revision 1. Thus, the support box design meets the requirements of ASME Code, Section III, Subsection NF, 1998 Edition, up to and including the 2000 Addenda. The support structure is classified as NF Class 1. Allowable stress limits of ASME Code NF-3220 for the Level A, B, C and D load combinations were calculated for the material used [ ] and it was found that the Level D loading condition governs the design. For design analysis, a three-dimensional finite element analysis (FEA) model was constructed using the general purpose FEA computer code ANSYS. Details of the methodology including the modeling techniques, boundary conditions and loading input are documented in the report APP-PH01-Z0C-007, "Finite Element Analysis of the RV Support Structure," Revision 0, May 26, 2010. The resulting stresses, including linearized primary membrane (P_m) and primary membrane plus bending ($P_m + P_b$) stress intensities, were computed from the FEA model. [ ] The resultant stress intensities for the controlling Level D load case [ ] were compared with the allowable stress intensities. The comparisons demonstrate that the Level D loading condition [ ] is less than the allowable. The staff performed audits of a supporting document on design specifications (APP-SS30-Z0-001) and a supporting document on FEA (APP-PH01-Z0C-007). These audits confirmed that the information in the supporting documents was consistent with the applicant's proposed design changes. Therefore, the staff finds the proposed design changes to be acceptable because they satisfy the ASME Code requirements.

In its September 9, 2010, letter, the applicant stated that the RPV support bottom and side wear plates are specified as Lubron wear plates fabricated by Lubron Bearing Systems. [ ] Detailed specifications of the wear plates used were provided by the applicant. Specifications of the bottom and side wear plates are provided in AP1000 RPV support design drawings APP-PH01-V2-211 (General Assembly) and APP-PH01-V2-212 (Component Details). During RPV installation, the bottom wear plate and interfacing thermal plate are assembled with a bluing process to insure a high degree of uniform contact (>75 percent) between mating surfaces. By design, these wear plates are good for up to 55.2 MPa (8,000 psi) bearing pressure and up to 593 °Celsius (C) (1,100 °Fahrenheit (F)). The actual bearing pressure for the AP1000 RPV support for the Level A service condition is approximately 15.9 MPa (2,300 psi), much less than the bearing capacity of 55.2 MPa (8,000 psi). The RPV support, its connection to the foundation, the foundation, and the wear plate connection to the support have all been designed for friction loads during normal plant heat-up and cool down thermal cycles. The staff's review also considered the potential for uneven settlement of the RPV during installation. Tier 1 Table 2.1.3-2 specifies the ITAAC for the reactor system, which includes the RPV. Several ITAAC exist to address this issue, and include acceptance criteria that a report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions. The staff finds that the proposed design changes are acceptable in terms of structural and tribological performance.

23.R.4 Conclusion

The staff concludes that these proposed design changes are acceptable because the proposed changes will adversely affect neither safety-related SSCs, nor the capability of the RPV support boxes to perform their intended function of supporting the RPV while allowing free lateral movement of the four legs due to thermal expansion. These design changes were evaluated with respect to conformance with the existing AP1000 licensing basis and found acceptable.

23.S Changes to the Passive Containment Cooling System

23.S.1 Description of Proposed Changes

The applicant has proposed changes to the shield building from the Revision 15 DCD to address additional external hazards and to simplify construction. The applicant assessed the impact of these changes to the PCS in APP-GW-GLR-096, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis," Revision 1, and submitted the report to the NRC in a letter dated August 6, 2010.

This report includes a description of the enhanced shield building design changes, a discussion of how these changes impact design basis test results, and evaluations of the limiting DBA and beyond design basis accidents (BDBA). Appendix A of APP-GW-GLR-096 includes proposed DCD changes. The changes that impact NUREG-1793 Sections 6.2.1, "Containment Functional Design"; 6.2.2, "Containment Heat Removal Systems"; and 6.2.3 "Shield Building Functional Design"; are as follows:

- Update containment response to reflect results of an analysis that incorporates the enhanced shield building design.

- Lower the required reactor decay heat limit for air only cooling.

- Specify the long-term makeup rates the PCS must simultaneously provide to containment and the SFP when the plant is refueling.

These proposed design changes also include revisions to several TS. In TS Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentation"; TS Table 3.3.5-1, "DAS Manual Controls"; and TS 3.6.7, "Passive Containment Cooling System (PCS) – Shutdown"; the Modes 5 and 6 applicability is revised to reflect the lower reactor decay heat limit for air only cooling mentioned above. In TS 3.7.9, "Fuel Storage Pool Makeup Water Sources," the notes associated with LCO 3.7.9, which list special plant conditions for each available makeup water source, are revised to account for changes to TS 3.6.7 regarding the availability of the passive containment cooling water storage tank (PCCWST) as a makeup water source. The cask loading pit (CLP) is added as a third makeup water source in addition to the PCCWST and the cask washdown pit. Two new surveillance requirements are also added to ensure readiness of water inventory from the PCCWST and the CLP, when needed.

In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. Proposed changes to Chapter 9 are evaluated in Section 9.1.3 of this report.

23.S.2 Regulatory Basis

The following Commission regulations are related to the evaluation of the enhanced shield building:

- GDC 16, "Containment Design," as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require

- GDC 38, "Containment Heat Removal," as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain them at acceptably low levels

- GDC 50, "Containment Design Basis," as it relates to demonstrating sufficient margin in accident analysis

- 10 CFR 52.47(c)(2), as it relates to design certification testing in support of a passive plant design

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS and the SFP design and operating information described in DCD Sections 6.2 and 9.1.3 respectively. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.S.3 Evaluation

One of the safety-related functions of the PCS is to provide air flow over the outside of the containment vessel by means of a natural circulation air flow path. Air enters through inlets located at the top of the shield building and then travels downward into the annulus between the inside of the shield building and the air baffle. At the bottom of the baffle wall, the air turns 180 degrees into the riser annulus formed by the baffle and the outside of the containment vessel. Air continues past the top of the containment vessel and exits through the shield building chimney.

In the Revision 15 design, air is admitted into the top of the shield building through 15 large, uniformly distributed openings, with fixed louvers and screens to prevent weather and wildlife from entering the shield building. In the modified design, a steel structure containing 29 uniformly distributed openings with fixed louvers and screens is added around the top outside of the shield building. The flow area in the steel structure is slightly higher than the flow area through the louvered and screened inlet openings in the previous shield building design. Air enters through these openings, collects in the plenum created by the structure and outside of the shield building, and then enters the shield building through 236 inlet ducts. These shield building inlets have a much lower flow area than the louvered and screened inlet openings in the previous shield building design. The enhanced shield building chimney is shorter than the original design; therefore, the buoyant driving head for the PCS flow is expected to decrease. In the enhanced design, two heavy steel grates are added within the chimney region to protect the containment shell from external hazards, which will increase the discharge flow resistance and reduce the discharge flow area.

The NRC approved the functional design of the containment systems for the AP1000 as described in NUREG-1793. The agency based its approval on data from a number of test facilities that were specifically designed to model processes and phenomena of the AP1000 containment design following a major piping rupture within the containment building. These data were used for the qualification of the WGOTHIC computer code, which was then used for the safety analysis of the AP1000. The test series included heat and mass transfer tests, wind tunnel tests, water distribution tests, integral system tests of the PCS, and air flow path characterization tests. All of these tests were to be within the range of the physical conditions predicted for the enhanced shield building design except for the air flow path characterization test which modeled the previous shield building. The flow path pressure drop is expected to increase in the new design; therefore, the applicant repeated the air flow path characterization test on apparatus specific to the revised shield building. The results were used to confirm the local annulus loss coefficients used in the revised WGOTHIC model.

From the phenomena identification and ranking table that was developed for the AP1000 in Westinghouse Commercial Atomic Power (WCAP)-15613, “AP1000 PIRT and Scaling Assessment,” issued February 2001, The applicant identified the high-ranked (most important) phenomena and processes for evaluation of the containment response following a LOCA or main steamline break (MSLB). These include condensation on the inside surface of the containment shell, conduction through the shell, and evaporation from the containment shell liquid film that flows down from the PCS water storage tank to the top outside of the containment building. Air flow through the containment shield building annulus is not listed with the high-ranked phenomena.

The staff agrees that air flow through the shield building need not be highly ranked following a LOCA or MSLB, because in these instances, the PCS will release water over the containment shell to provide evaporative cooling, and evaporative cooling is a much more significant heat removal mechanism than air flow convection. Analyses by the applicant and the staff illustrate this conclusion. APP-GW-GLR-096 describes the Westinghouse analysis based on the WGOTHIC evaluation model that was approved by the staff for the Revision 15 DCD analysis and subsequently modified to incorporate the proposed wet bulb temperature increase to 30 °C (86.1 °F) discussed in Section 6.2.1.1.1 of the DCD along with the pressurizer room changes discussed in Section 6.2.1.2 of the DCD.

To address the enhanced shield building design, the applicant further revised the model to include the addition of the PCS air inlet structure, reduction of the flow areas in the shield building inlet and exit, reduction of the shield building chimney height, and increase to the air flow path resistance. The applicant ran the design basis LOCA and MSLB events with the revised model and the results demonstrated that even though the natural circulation air flow decreased, the effect on containment pressure and temperature was insignificant. Additionally, the pressure at 24 hours following a large cold leg break LOCA was unchanged. The staff reviewed the detailed modeling changes during audits on April 21, 2010 and September 3, 2010. The subject of the audits was APP-SSAR-GSC-746, “Containment

Response Analysis for the AP1000 Shield Building Design Change." The staff found the estimates on loss coefficients, which were made prior to completion of the air flow path characterization test, to be reasonable because they bounded predictions [ ] for a thick orifice. Furthermore, when pressure drops from the air flow path characterization test became available, these estimated loss coefficients were demonstrated to be conservatively high.

The staff performed a confirmatory analysis using the CONTAIN computer code with an AP1000 model developed by the staff during the DCD review. The shield building changes were incorporated and used to evaluate a large cold-leg break LOCA, which is the peak pressure DBA. The results were consistent with the Westinghouse evaluation; the PCS air flow decrease had a negligible impact on peak pressure, peak temperature, and containment pressure after 24 hours.

The applicant proposed changes to incorporate the results of its analysis into the DCD tables summarizing postulated accident values, DCD Tier 2, Tables 6.2.1.1-1 and 6.2.1.1-3. The changes to the LOCA and MSLB results were found acceptable because they are consistent with the enhanced shield building design. Changes made to the external pressure results are evaluated in Section 23.W of this report.

The natural circulation air flow is reduced in the enhanced shield building design; therefore, the amount of heat that can be removed during air only PCS operation will also be reduced. As a result, the applicant proposed lowering the reactor decay heat limit for air only PCS operation from 9 megawatts thermal (MWt) to 6 MWt. APP-GE-GLR-096 describes the supporting WGOTHIC analysis demonstrating the containment pressure could be maintained below the design value of 407 kPa (59 psig) for seven days with no PCS water release (and subsequent evaporative cooling) assuming an initial decay heat rate of 6 MWt.

The staff reviewed the analysis basis during its July 27, 2010, audit of APP-SSAR-GSC-749, "AP1000 Dry PCS Heat Removal Capability." The evaluation model was the previously described LOCA model modified to turn off the PCS water, replace the accident mass and energy forcing functions with decay heat input to the IRWST, and adjust the air flow path loss coefficients to reflect a 30 percent increase over the air flow path characterization test results. The staff finds the loss coefficient values reasonable because they are based on physical measurements with added margin to bound uncertainties in the test results.

The staff ran a confirmatory CONTAIN analysis using the previously described LOCA model modified for no PCS water flow and an initial decay heat rate of 6 MWt.

The results were consistent with the applicant's evaluation; the air flow across the containment shell decreased and the containment pressure increased compared to the original shield building, but containment pressure remained below the design value for the run duration of seven days.

In an additional study, the applicant re-evaluated the containment pressure resulting from the BDBA of a prolonged loss of offsite power concurrent with a complete loss of the passive containment cooling water. For this unlikely event, the PRHR HX transfers reactor decay heat and the system sensible heat into the IRWST pool. When the water in the IRWST is heated to boiling, the steam that is released condenses on the containment internal structures. As the internal structures are heated, the containment pressure rises and heat is transferred through the containment shell to the air traveling up through the shield building annulus. The applicant performed the analysis with a “best estimate” WGOTHIC model. The results showed that even with the reduced natural circulation air flow associated with the enhanced containment building, it will take more than 24 hours for the containment pressure to reach the maximum pressure capability limit of 889 kPa (129 psig) defined in DCD Section 3.8.2. Therefore, the enhanced shield building meets the DCD Section 19.34.2.6 statement that, with air-only cooling, containment failure is predicted to occur more than 24 hours after accident initiation. The staff reviewed the analysis basis during its July 27, 2010, audit of APP-SSAR-GSC-749. The assumptions for the “best estimate” BDBA evaluation model, which included changes to initial temperatures, heat transfer coefficients, and loss coefficients, were found to be reasonable and consistent with evaluations of beyond design basis events.

The PCS has a design commitment to provide containment cooling and SFP makeup simultaneously from post-72 hours to seven days after DBA initiation at the minimum flow rates specified in DCD Tier 2, Table 6.2.2-1. The applicant recognized that the proposed reduction to the maximum reactor decay heat limit for air only containment cooling created a scenario whereby the currently specified minimum SFP supply of 132.5 Lpm (35 gpm) would not be adequate to maintain coverage of the fuel in the SFP. During refueling, when the full core is split such that the reactor decay heat is greater than 6 MWt and the SFP decay heat is less than 7.2 MWt, the PCS water is reserved for containment cooling for the first 72 hours and cannot be used for SFP makeup until after this time. The applicant determined a DBA at this plant condition would require a minimum of 189.3 Lpm (50 gpm) SFP supply. Because the total PCS flow is limited, an increase in the SFP supply requires a reduction to the currently specified containment make-up rate of 378.5 Lpm (100 gpm). The 378.5 Lpm (100 gpm) is based on containment cooling requirements post-72 hours after a limiting DBA occurring at full power. The necessary flow following a DBA during refueling will always be less than this because the reactor must be shut down for 100 hours prior to the start of refueling to provide time for the RCS to cool down and depressurize. The applicant reduced the required flow rate for containment cooling during refueling to 302.8 Lpm (80 gpm). The supporting analysis, APP-SSAR-GSC-750, “WGOTHIC Validation of Post-72 Hour Containment Cooling Flow Rates for Accident Scenarios after Refueling” was audited by the staff on July 30, 2010.

The evaluation considered a loss of power event coincident with the start of refueling, modeled by adding decay heat representative of 100 hours after shutdown to the IRWST to represent the PRHR HX system. The model assumed full PCS flow for the first 72 hours and 302.8 Lpm (80 gpm) flow thereafter. The results demonstrated that the containment pressure remained well below the design limit of 407 kPa (59 psig) for seven days. The staff found the modeling assumptions to be consistent with design basis analysis and the significant margin in the results provided further assurance that this is an acceptable change.

With respect to proposed changes to TS Sections 3.3.2, 3.3.5, 3.6.7 and 3.7.9 and their associated Bases, the staff finds these changes acceptable because they reflect the PCS and the SFP cooling system design and operating information described in DCD Sections 6.2 and 9.1.3, respectively. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.S.4 Conclusion

The staff’s review concludes that the design changes described above are acceptable and compliant with GDC 16, GDC 38, GDC 50 and 10 CFR 52.47(c)(2).

23.T Changes to the Main Control Room Emergency Habitability System

23.T.1 Description of Proposed Changes

In letters dated July 29, 2010, and September 22, 2010, the applicant has proposed changes to the minimum amount of stored compressed air in the MCR VES emergency air storage tanks. To increase the margin to the control room operator dose limits and to expand the site dispersion factors that are permitted for the AP1000 design, a passive filtration subsystem design was added to the MCR VES to filter potential contaminated in-leakage. The subsystem incorporates an eductor, which uses the VES compressed air flow to induce recirculation of MCR air through a filtration unit. The performance of the added subsystem allows for 0.42 cubic meter per minute (m^3/m) (15 cubic feet per minute (cfm) of unfiltered in-leakage while maintaining operator dose below 50 millisievert (mSv) (5 roentgen equivalent man (rem)) total effective dose equivalent (TEDE) required by GDC 19. With the addition of the subsystem, the VES provides a filtration unit to capture potential contaminated air that may leak into the MCR envelope. The applicant conducted testing of the passive filtration subsystem at the Westinghouse Waltz Mill facility. The testing was confirmatory testing to show the performance characteristics of the added passive filtration design and to collect data on the performance of the eductor itself. The passive filter train, utilizing the eductor as well as the technical specifications and ITAAC were reviewed and approved in Sections 6.4 and 9.4.1 of this report. However, after the Chapter 6 and Chapter 9 SERs were issued, the applicant identified the need to adjust the TS on quantity of compressed air needed for 72 hours of continuous operation. The quantity of air needed to support 72 hours of operation increased because the pressure regulating valve minimum required operating inlet pressure needed to increase to ensure that the set outlet pressure could be maintained during the duration of system operation.

The minimum amount of required stored compressed air is changed from 8895 standard cubic meter (m^3) (314,132 standard cubic foot (scf)) to 9276 standard m^3 (327,574 scf. Filling the tanks with 9276 standard m^3 (327,574 scf of air will ensure that the tanks are capable of providing 119 standard cubic meters per hour (scmh) (70 standard cubic foot per minute (scfm)) of air for 72 hours. Operability is determined based on the amount of compressed air stored in the tanks as determined from tank pressure and storage room temperature. The new volume of air is provided in both Tier 1 and Tier 2, including the TS. The relationship between tank pressure, room temperature and volume is provided in the TS Bases.

These proposed design changes also modified the instrumentation representation in DCD Figure 6.4-2 (Sheet 2 of 2) to better represent the instrumentation used in the design. The figure changed a flow instrument from an orifice plate with a differential pressure sensor to a thermal dispersion mass flow transmitter.

23.T.2 Regulatory Basis

The applicable regulations for the control room habitability system aspects of these design changes are detailed in NUREG-0800 Section 6.4, “Control Room Habitability System.” For the changes proposed by the applicant, the following are relevant.

- 10 CFR Part 50, Appendix A, GDC 19

- 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS and Bases reflect the VES design and operating information described in DCD Section 6.4. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance specified in NUREG-0800 Chapter 16.

23.T.3 Evaluation

The proposed design changes were reviewed for compliance with the applicable regulations. This DCP raises the minimum amount of stored compressed air in the VES emergency air storage tanks to ensure the supply of 72 hours of air at the maximum flow rate of 70 scfm. The mission time for the passive safety-related system is 72 hours based on the Commission's policy on passive systems. The design basis requires that 65±5 scfm flow be provided for all DBAs. As a result, there needs to be enough air to provide 72 hours of flow at the maximum flow of 119 scmh (70 scfm).

The change is necessary to accommodate a 1379 kPa (200 psig) pressure regulator minimum inlet pressure instead of 689 kPa (100 psig) prior to the passive filter train. This is done to ensure that downstream pressure and flow are maintained within the required tolerance. The applicant has demonstrated that the new volume of air is adequate to accomplish the design basis functions.

The staff audited APP-VES-M3C-005, "VES Minimum Pressure Calculations," Revision 1. The calculation determines the amount of breathable air required during 72 hours of VES operation. Basic heat transfer equations and air properties were used to determine heat transfer coefficients and transient heat loss from concrete to outside air and the tank package. Cooldown analysis for station blackout during extreme winter weather were used to determine VES tanks pressure decline due to temperature cooldown and air delivery to MCR. Assuming cold conditions is conservative because it minimizes the volumetric flow. The staff has found that the calculations demonstrate there is an adequate amount of gas to maintain pressure above 1379 kPa (200 psig) at the inlet of the pressure regulator. The calculations showed that for the limiting case, there was not much margin. However, because the calculations were done for the limiting cases, small margin is acceptable.

In Chapter 6 of this report, the staff accepted the passive filter train. The staff came to this conclusion, in part, based on there being adequate ITAAC and Technical Specifications to demonstrate that 65±5 scfm flow is provided from the canisters to the eductor and that 600 scfm of control room air would be drawn through the filter train. The COL holder would need to demonstrate prior to plant operation and periodically thereafter that the system would accomplish the design basis functions. Because the passive eductor-driven filter train was new to the nuclear industry, the staff requested that testing be done to prove this design concept was capable of being operated successfully. Prior to the issuance of the Chapter 6 SER, the applicant completed testing to demonstrate that the system was capable of meeting the performance requirements.

When these proposed design changes were submitted by the applicant, the staff wanted to make sure the testing still demonstrated the system was capable of being operated successfully. As such, the staff also audited TS-SEE-111-09-03, "AP1000 VES Air Filtration Test Specification." The testing performed by the applicant had three objectives. The first was to demonstrate 60 scfm is capable of inducing 600 scfm, the second was to determine whether a feed flow rate higher than the design duct flow rate would damage system components, and the third was to demonstrate the system can be operated below maximum allowable noise levels defined in NUREG-0700, “Human-System Interface Design Review Guidelines.” The maximum allowable noise is 65 dB(A).

The staff observed that the testing did demonstrate the system could be built with the necessary performance characteristics. The staff also observed that the test results showed that the combined flow is sensitive to back pressure, feed pressure and feed flow. For example, for the same feed pressure, an increase in backpressure by 0.5 inch of water will reduce combined flow by approximately 170 scmh (100 scfm). Additionally, at the same feed flow an increase in the feed pressure by 34 kPa (5 psi) will reduce combined flow by approximately 170 scmh (100 scfm). As a result, the staff notes that the parameters influencing the induced flow, for example line losses between the regulator and the eductor, will need to be carefully controlled by the COL holder and that the required 1020 scmh (600 scfm) induced flow may not be satisfied at all conditions. The COL holder has the responsibility to ensure that at least 1020 scmh (600 scfm) will be induced by a feed flow rate of at least 1020 scmh (60 scfm).

The two safety-related flow rates will be demonstrated by ITAAC. Additionally, Tier 1 requires the following be demonstrated:

- The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.
- The airflow rate from VES is at least 60 scfm and not more than 70 scfm.
- The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.
- The air flow rate at the outlet of the MCR passive filtration system is at least 600 cfm greater than the flow measured by VES-FT003A/B.
- The noise at the operator station is limited to 65 dB(A).

The staff has concluded that with the ITAAC to demonstrate the capacity of the system design the system will meet the requirements of GDC 19. Additionally, the ITAAC are sufficient to show the as-built plant will function and, as a result, 10 CFR 52.47(b)(1) is satisfied.

With regard to the change in instrumentation, the staff finds that either type of instrument can acceptably measure flow. The instrument is safety-related and subject to the quality assurance requirements. As a result, the staff finds the instrumentation change acceptable as well.

With respect to proposed changes to TS 3.7.6 and its associated bases, the applicant stated that both tank pressure and room temperature are used to determine the acceptable minimum storage capacity of the air tanks in term of standard cubic feet (scf). The acceptance criteria are presented in new Figures B 3.7.6-1 and B 3.7.6-2 for use in the verification of the minimum storage capacity of the air tanks specified in Required Action D.1 and SR 3.7.6.2, respectively. The staff finds these changes acceptable because they reflect the VES design and operating information described in DCD Section 6.4,

23.T.4 Conclusion

The staff finds that there will be an adequate amount of air for the extreme winter conditions for 72 hours of VES operation and to maintain pressure above 1379 kPa (200 psig) at the inlet of the pressure regulator. The staff has concluded that the proposed design change complies with GDC 19 and the acceptance criteria specified in Section 6.4 of NUREG-0800. Lastly, the staff finds that the ITAAC are sufficient to demonstrate that the system when built will accomplish the safety function.

23.U Changes to Main Steam Isolation Valve Subcompartment

23.U.1 Description of Proposed Changes

In letters dated, August 12, 2010, and September 30, 2010, the applicant proposed design changes that make the vent paths associated with the main steam isolation valve (MSIV) subcompartments larger. The applicant also proposed to change the content of the pipe hazards analysis report described in Tier 2 DCD Section 3.6.2.5 and to remove tables in Section 6.2 of the DCD that report mass and energy releases and compartment differential pressures outside of containment.

NUREG-0800 Section 6.2.1.2 describes subcompartment analyses inside of containment. Additionally, NUREG-0800 Section 6.2.1.4 describes mass and energy release from secondary side breaks inside of containment. In Chapter 3 of NUREG-0800, high energy pipe hazards outside of containment are described. In the certified design, subcompartment analyses and mass and energy data for pipe breaks outside of containment are included in Chapter 6. This is not typical and not consistent with NUREG-0800 or RG 1.206, “Combined License Applications for Nuclear Power Plants.” Additionally, there is a COL holder item in FSAR Chapter 3 that requires the COL to perform a pipe hazards analysis. As a result, there is some confusion in the certified design. In DCD Section 6.2.1.2 entitled, “Containment Subcompartment Analysis,” there are analyses for subcompartments outside of containment. Additionally, in DCD Chapter 3 there is the requirement for another pipe hazards analysis to be performed by the COL holder.

The applicant identified that the rupture of a feedwater pipe may produce more limiting results than the main steam line break that was described in the DCD; therefore, larger vent paths were needed in the main steam valve rooms.

The applicant has proposed to increase the size of the vent paths in the roof of the Auxiliary Building to provide larger vent paths for high-energy hazards. The proposed design changes also modify the structural attachments at the Auxiliary Building roof. The applicant identified a pipe hazard that releases more energy than is currently considered in the hazards analysis. The vent paths are described in Chapter 3. Specifically the applicant has made changes in the doghouse structures on the roof of the Auxiliary Building. Each doghouse structure has a blowout panel and a louvered vent. The blowout panel is 3.04 m (10 ft) by 3.04 m (10 ft) and the louvered vent is 1.82 m (6 ft) by 1.82 m (6 ft).

The applicant has clarified Section 3.6.2.5 to explicitly include subcompartment pressurization of these compartments outside of containment in the pipe hazards analysis report. To eliminate ambiguity in the DCD, the applicant has removed the subcompartment analyses and mass and energy tables for pipe hazards outside of containment from Chapter 6 of the DCD.

23.U.2 Regulatory Basis

For the Chapter 6 changes, the applicable regulations for the containment systems aspects of this design change are detailed in NUREG-0800 Section 6.2.1.2, “Subcompartment Analysis,” and include the following:

- 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases”

- 10 CFR Part 50, Appendix A, GDC 50

However, because the subject subcompartments are not inside the containment, GDC 50 is not applicable.

For the Chapter 3 changes, the applicable regulations for the protection against pipe rupture are detailed in NUREG-0800 Sections 3.6.1, “Plant Design For Protection Against Postulated Piping Failures In Fluid Systems Outside Containment”; 3.6.2, “Determination Of Rupture Locations And Dynamic Effects Associated With The Postulated Rupture Of Piping”; and in Branch Technical Position (BTP) 3-3, “Protection Against Postulated Piping Failures In Fluid Systems Outside Containment.”

23.U.3 Evaluation

The proposed design changes were reviewed for compliance with GDC 4, which requires that structures withstand the effects of high energy hazards. The design of internal compartments must accommodate the effects of, and be compatible with, the environmental conditions associated with postulated accidents or high energy hazards. The internal compartments shall be appropriately protected against dynamic effects.

The proposed design change of increasing the size of the vents improves the room’s or compartment’s ability to relieve pressure if a high energy pipe fails in the compartment. Confirmation that the final as-built design is in compliance with the requirements is the responsibility of the COL holder. This responsibility is clearly described in the COL holder item in DCD Section 3.6.4.1. Section 3.6.2.5 of the DCD outlines the content of the pipe hazards analysis report. The applicant proposed to change Section 3.6.2.5 to include the following statement:

> Evaluate compartment pressurization in the break exclusion zones in the vicinity of containment penetrations due to 1.0 square foot breaks in the main steam and feedwater lines.

The proposed change clarifies that the compartment pressurization in the break exclusion zones needs to be evaluated for 1.0 square foot breaks in the main steam and feedwater lines. The proposed design change is acceptable because it improves the ability of the facility to withstand high-energy pipe breaks. The change to Section 3.6.2.5 is acceptable because it clarifies the content of the high-energy line break analysis necessary. A COL holder item for high-energy line break analysis is an appropriate approach to demonstrate compliance with GDC 4 because much of the pipe hazards analysis is site-specific in nature.

The staff also finds the removal of the peak differential pressures and mass and energy tables from DCD Chapter 6.2 acceptable. Section 6.2 is dedicated solely to containment issues.

Neither NUREG-0800 nor RG 1.206 recommends these items be included in Chapter 6. Outside containment subcompartments are much more appropriate in Chapter 3 under the COL holder item. As a result, the staff finds this change acceptable as well.

The DCD Tier 2 Section 3.8.4.3.1.4 specifies that a differential pressure of 41 kPa (6 psid) be the design limit for subcompartment pressurization in the MSIV rooms. The design finalization shows that a main feedwater line pipe rupture without adequate venting will cause the subcompartment pressurization to exceed the 41 kPa (6 psid) design pressure limit. Accordingly, the applicant decided to enlarge the venting area of the roof of the Auxiliary Building in order to meet the requirements of the DCD. The applicant provided the modified steam vent design with the proposed design changes. In addition, the applicant committed to finalize the design of the two doghouse structures on the roof of the Auxiliary Building in accordance with the design procedure of the critical sections as defined in DCD Section 3.8, its appendices, and in technical report (TR)-57, APP-GW-GLR-045, "Nuclear Island: Evaluation of Critical Sections."

During the staff's evaluation of these proposed design changes, the following additional information was requested from the applicant:

- Engineering drawings of the doghouse structure at the roof, detailing the venting assemblies, blow-out panels, louvered vents, and connections to the Auxiliary Building roof.

- Stress evaluation on the structures, including applied design-basis loading and the resulting maximum stress.

- The basis, including acceptance criteria and performance requirements, upon which the proposed new design is determined to be acceptable.

The applicant's September 30, 2010, response stated that the design will be finalized using the same methodology, specifications for load combinations, and safety factors that are specified in DCD Section 3.8. Because the final design will be consistent with the design procedure of the critical sections as defined in DCD Section 3.8, its appendices, and in TR-57, the staff finds the change to the vent area to be acceptable.

23.U.4 Conclusion

On the basis of its review of Sections 3.6.2 and 6.2, the staff finds the proposed design changes acceptable. The approach chosen by the applicant will ensure compliance with GDC 4.

The staff's review concludes that the proposed design changes are acceptable because they will not adversely affect safety-related SSCs or the capability of the MSIV depressurization subcompartments to perform their intended functions of pressure relief for the postulated high-energy line break.

23.V Changes to the Component Cooling Water System

23.V.1 Description of Proposed Changes

A leak or tube rupture in the RCP external heat exchanger (EHX) would not result in over-pressurization of piping outside containment, since the pressure in the CCS is controlled by the system's atmospheric surge tank. Such an event could result in a reactor trip and a nonisolable flow path from the RCS through the CCS piping and the surge tank vent to the turbine building (interfacing system LOCA). The applicant has proposed design changes intended to add a safety-related means to isolate this potential flow path. The proposed design changes would provide automatic, safety-related isolation of a LOCA caused by the rupture of one of the RCP EHX tubes. This automatic isolation would prevent discharge of reactor coolant to the turbine building through the CCS surge tank vent and would limit offsite doses to values below those already found acceptable in the event of a small break LOCA, as stipulated in 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors."

In letters dated July 28, 2010; September 3, 2010; September 29, 2010; and October 18, 2010, the applicant proposed design changes to the CCS. The proposed design changes relate to the following:

- Instrumentation and Controls – Containment isolation:

  The applicant proposed to modify the closure logic for CCS motor-operated containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 to add a requirement to close on generation of the RCP bearing water high temperature pump trip signal. This modification would add a new isolation signal to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetration and Isolation Valves."

  A closure signal to the component cooling system containment isolation valves is derived from a coincidence of two of the four divisions of high RCP bearing water temperature for any RCP. The high temperature setpoint and dynamic compensation are the same as used in the high RCP bearing water temperature RCP trip (Section 7.3.1.2.5, Condition 6), but with the inclusion of preset time delay.

- Instrumentation and Controls – CCS:

  The applicant proposed to remove the automatic isolation of the CCS RCP HX outlet isolation valves (CCS-PL-V256A/B/C/D) to close on high deviation between inlet and outlet flows. Simultaneous flow deviations in both the inlet and outlet lines would generate a flow deviation alarm and not isolate these valves. This alarm would be indicative of RCS leak conditions and would alert plant operators to close the valve on the cooling water outlet line on each RCP to prevent reactor coolant flow throughout the CCS. Both the flow signals and the isolation valves are nonsafety-related.

- CCS:

  The applicant proposed to install a 10.16 cm (4 in) x 15.24 cm (6 in) safety-class relief valve, designated CCS-PL-V270 and CCS-PL-V271, respectively, on each of the 25.40-cm (10-in) CCS supply and return lines (total of two relief valves), just inside the innermost containment isolation valves (CCS-PL-V201 and CCS-PL-V207). In addition, the applicant proposed to change the safety class of the section of line between the innermost containment isolation valves and the Appendix J test valves (CCS-PL-V214 and CCS-PL-V216) from Class ‘0’ to Class ‘C’ to ensure that the relief valves are installed as ASME safety-class piping.

- Technical Specifications:

  The applicant proposed to add an RCP bearing water temperature high trip function to TS Table 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” for closure of the CCS containment isolation valves.

These proposed design changes include revisions to the Tier 2 DCD Sections 3.2, 3.9, 3.11, 5.2, 6.2, 7.2, 7.3, 9.2.2, and Chapter 16.

23.V.2 Regulatory Basis

- Instrumentation and Controls – Containment isolation

  The regulatory basis for evaluating the changes to the AP1000 closure logic for CCS motor-operated containment isolation valves is documented in Chapter 7 of NUREG-1793. Review of these proposed design changes is also based on the requirements in 10 CFR 50.55a(h), 10 CFR 52.47, and GDC 20 and GDC 21 of 10 CFR Part 50, Appendix A.

  The applicable regulations for the containment systems aspects of this design change are detailed in NUREG-0800 Section 6.2.4, “Containment Isolation System,” and include the following:

  - GDC 16 requires that the containment isolation system allow the normal or emergency passage of fluids through the containment boundary while preserving the capability of the boundary to prevent or limit the escape of fission products from postulated accidents.

  - GDC 54, "Piping Systems Penetrating Containment," requires that the containment isolation system valves in piping systems that penetrate the containment be designed to close reliably under accident conditions and prevent the uncontrolled release of radioactive materials.

- Instrumentation and Controls – CCS:

  The regulatory basis for evaluating the proposed instrumentation and controls changes is documented in Chapter 7 of NUREG-1793. Both the flow signals and the isolation valves for the CCS RCP HX outlet isolation valves are nonsafety-related and are not relied upon to perform any safety functions. However, they are part of the CCS, which is considered to be important to safety because it supports the normal (defense-in-depth) capability of transferring heat from various plant components and also removing reactor system and spent fuel decay heat. Reviews of the changes are based on meeting the relevant requirements of 10 CFR 50.55a(h), 10 CFR 52.47, GDC 13, and GDC 19 of 10 CFR Part 50, Appendix A.

- CCS:

  The regulatory basis for evaluating the CCS is documented in Section 9.2.2 of NUREG-1793. While the CCS is a nonsafety-related system, it is considered to be important to safety because it supports the normal (defense-in-depth) capability of removing reactor and spent fuel decay heat. It is part of the first line of defense for reducing challenges to passive safety systems in the event of transients and plant upsets, and its cooling function is important for reducing shutdown risk when the RCS is open (e.g., mid-loop condition). The risk importance of the CCS makes it subject to RTNSS in accordance with the Commission's policy for passive reactor plant designs. The staff's evaluation of the changes that are proposed focused primarily on confirming that the changes could not adversely affect safety-related SSCs or those that satisfy the criteria for RTNSS; the capability of the CCS to perform its defense-in-depth and RTNSS functions; and the adequacy of ITAAC, test program specifications, and availability controls that have been established for the CCS. The proposed changes were evaluated using the guidance provided in NUREG-0800 Section 9.2.2, "Reactor Auxiliary Cooling Water System – Revision 4, March 2007," as it pertains to these considerations. Acceptability was based on conformance with the existing AP1000 licensing basis, the guidance specified in NUREG-0800 Section 9.2.2 (as applicable), and the Commission's policy with respect to RTNSS.

  The following regulatory guidance is also applicable to the proposed design changes:

  - RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"

  - RG 1.29

  - SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," related to overpressurization of low-pressure piping systems due to RCS boundary isolation failure could result in rupture of the low-pressure piping outside containment. This could result in a core melt accident with an energetic release outside the containment building, potentially causing a significant offsite radiation release.

- Technical Specifications:

  The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the CCS isolation function design and operating information described in DCD Sections 5.2, 7.3, and 9.2, respectively. The proposed changes were evaluated using the guidance provided in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.V.3 Evaluation

During the staff's evaluation of these proposed design changes, additional information was requested from the applicant. In a letter dated September 3, 2010, the applicant responded to the request for additional information. The staff reviewed the applicant's responses and came to the following conclusions.

- The staff determined that the CCS piping system is adequately protected from over-pressurization due to a postulated RCP external heat exchanger tube rupture with the addition of two new relief valves, one near each CCS containment penetration. Both relief valves are designed to ASME Code Section III, Class 3 and have been added to a markup of Table 3.9-16, "Valve In-service Test Requirements," with a test frequency of at least once per 10 years. The failure of a spring-operated safety valve to open on a high pressure condition is excluded as a single active failure; however, two relief valves will see the over-pressurization event since there is no check valve between the cooling water line serving the reactor coolant drain tank. Since each valve has sufficient capacity to prevent system overpressure for the largest expected discharge from the heat exchanger tube rupture event, the staff finds the CCS piping system is adequately protected from over-pressurization. Therefore, the staff finds the proposed design changes to be acceptable.

- The proposed design changes were reviewed for compliance with GDC 16 and GDC 54 pertaining to containment isolation and containment integrity. The proposed design changes modify the closure logic for the motor operated CCS containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 by adding a requirement to close on the generation of a RCP bearing water high temperature signal. This modification adds an additional isolation signal to these valves, and has been included in AP1000 DCD, Tier 2, Table 6.2.3-1, "Containment Mechanical Penetration and Isolation Valves."

  Containment isolation valves are required to close reliably under accident conditions and maintain their integrity to prevent the uncontrolled release of radioactive materials from containment. RCP bearing high water temperature would result from a rupture of one of the U-tubes in the RCP-EHX, which is cooled on the shell side by the CCS. The tube rupture would result in both increased temperature and increased pressure in the CCS. Both GDC 16 and GDC 54 require the containment isolation valves to close to prevent the uncontrolled release of radioactive materials from containment. Therefore, the containment isolation valves must operate at the pressure and temperature conditions in the CCS generated by the RCS tube rupture. Additionally, the containment isolation valves must maintain their integrity and remain leak tight when closed.

  Prior to a rupture, the tube side temperature and pressure in the RCP-EHX are approximately 71 °C (160 °F) and 15.5 MPa (2250 pounds per square inch absolute (psia)); whereas, the temperature and pressure on the shell side of the heat exchanger are approximately 36.6° C (98 °F) and 793 kPa (115 psia). The CCS is a 1379 kPa (200 psig) system, including the containment isolation valves and their included piping. Following a tube rupture the CCS pressure and temperature in the containment would increase significantly. The following conditions must be satisfied:

- The 1379 kPa (200 psig) design containment isolation valves CCS PL-V207, PL-V208, PL-V200, and PL–V201 must be able to close against the increased pressure due to the RCP-EHX tube rupture.

- The 1379 kPa (200 psig) design containment isolation check valve CCS-PL-V201 must maintain its integrity at the increased pressure and temperature due to the RCS-EHX tube rupture and remain closed.

- The new safety-class relief valves, CCS-PL-V270 and CCS-PL-V271, must operate under the conditions of flashing fluid at the maximum fluid temperature produced by the RCP-EHX tube rupture event.

The applicant proposed to add a single safety-class relief valve on each of the 25.4 cm (10 in) CCS supply and return lines respectively just inside the innermost containment isolation valves, CCS-PL-V201 and CCS-PL-V207. The safety-class relief valves, with a set pressure of 1379 kPa (200 psig), are intended to limit the CCS pressure at the containment penetrations to the design conditions for a 1379 kPa (200 psig) system.

The applicant performed an analysis, AP1000/ANSALDO RELAP calculation APP-CCS-M3C-164, Revision 0, September 24, 2010, to simulate the occurrence of a postulated double ended tube break in the RCP-EHX in order to evaluate CCS transient and steady state conditions of temperature and pressure at the relief and containment isolation valves.

The results of the analysis for Case 1, which postulates both safety class relief valves providing overpressure protection and opening at the set pressure of 1379 kPa (200 psig), demonstrate the following:

- CCS-PL-V201 reaches a maximum pressure of 1482 kPa (215 psia), a maximum temperature of approximately 193.3 °C (380 °F), and reaches a steady state condition of approximately 1241 kPa (180 psia) and 189 °C (372 °F) at 1000 seconds.

- CCS-PL-V207 reaches a maximum pressure of 1489 kPa (216 psia), a maximum temperature of approximately 193.3 °C (380 °F), and reaches a steady state condition of approximately 1241 kPa (180 psia) and 189.4 °C (373 °F) at 1000 seconds.

The results of the analysis for Case 2, which postulates a single relief valve, V270, opening upon reaching the set pressure of 1480 kPa (200 psig), and a single failure of relief valve V271 to open, demonstrate similar results of temperature and pressure, but reach steady state conditions at 400 seconds.

The AP1000 piping class sheets and standard details, APP-PLO2-Z0-001, Revision 5, p. 120, provide the design pressure and temperature envelope for class JCB and JCC piping and the CCS containment isolation and safety-related relief valves, summarized as follows;

0-37.7 °C (32-100 °F)	65.5 °C (150 °F)	148.8 °C (300 °F)
1965 kPa	1862 kPa	1586 kPa
285 psig	270 psig	230 psig

Except for the CCS temperature of ~193.3 °C (~380 °F), the transient and steady state conditions are well within the pressure and temperature design envelope of the containment isolation valves and piping. In a subsequent revision to the valve and piping data sheets, "AP1000 Piping Class Sheets and Standard Details", APP-PLO2-Z0-001, Revision 5, the applicant made appropriate changes which resolves this issue.

A RELAP calculation by staff confirmed the above transient results from the applicant's analysis, which was submitted in a letter dated October 18, 2010. The staff concludes that the containment isolation valves would be able to close at these pressure and temperature conditions, and maintain their integrity following a tube rupture in the RCP-EHX.

Each of the four CCS containment isolation valves will be explicitly identified in DCD Tier 2, Table 3.9-12 (Sheet 1 of 7) as having both containment isolation and accident mitigation functions. The new safety-related relief valves, V270 and V271, will be explicitly identified as having an accident mitigation function. The conditions applicable for accident mitigation will be added to the valve data sheets as notes pertaining to the specific fluid temperature and pressure conditions for which the valves must remain operable and intact. In a subsequent revision to the valve and piping data sheets, "AP1000 Piping Class Sheets and Standard Details", APP-PLO2-Z0-001, Revision 5, the applicant made appropriate changes which resolves this issue.

RG 1.26 recommends that portions of cooling water systems important to safety and which are designed for the functioning of components important to safety be designed to Quality Group C standards. For piping and valves the applicable standard is ASME Section III, Class 3.

The applicant proposed to install two additional relief valves, CCS-PL-V270 and CCS-PL-V271, one on each of the 25.4-cm (10-in) CCS supply and return lines, just inside the innermost containment isolation valves, CCS-PL-V201 and CCS-PL-V207, respectively. These relief valves are intended to limit the CCS pressure at the containment penetrations to approximately 1379 kPa (200 psig). Since these relief valves are provided to protect the safety-related CCS containment isolation valves and piping, which are designed to Quality Group B standards, they will be designed to Quality Group C criteria, or ASME Code Section III, Class 3. The staff has evaluated this designation, which will be added to the DCD Tier 2, Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment," and concludes that it meets the recommendations of RG 1.26.

SSCs of a nuclear power plant designated as seismic Category I and designed to withstand the effects of an SSE and remain functional include components affecting the safety-related function of the primary containment, in accordance with RG 1.29. The relief valves, CCS-PL-V270 and -V271, are provided to protect the operability and long-term integrity of the CCS 25.4-cm (10-in) containment penetrations. Therefore, they will be designed to seismic Category I criteria. The staff has evaluated this designation and concurs that it meets the guidance of RG 1.29. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

For advanced reactor design, the staff stated its position regarding intersystem LOCA (ISLOCA) protection in SECY-90-016. The staff stated that advanced light-water reactor (ALWR) designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure.

The CCS is a low-pressure system design with an ultimate rupture strength (URS) below RCS operating pressure. Overpressurization of the CCS could occur following a tube rupture in the RCP-EHX. A relief valve has been added to each of the CCS supply and return headers inside the containment isolation valves in containment to prevent overpressurizing the CCS. The adequacy of pressure relief in protecting the operability and integrity of the containment isolation of CCS has been addressed above and found acceptable. Therefore, the staff finds CCS overpressure protection by the two safety-related relief valves meets the intention of the guidance for ISLOCA.

- Since DCD Tier 1 testing of the CCS-PL-V200 and V201 valves is adequately described in Table 2.2.1-1, the staff determined no other Tier 1 changes or ITAAC are required. DCD Tier 2, Section 14.2.9.2.5 adequately addresses CCS testing, which includes proper operations of controls, instrumentation, actuation signals and interlocks. Furthermore, DCD Tier 2, Section 14.2.9.1.10, “Containment Isolation and Leak Rate Testing,” addresses testing for proper operation of safety-related containment isolation valves listed in Table 6.2.3-1. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

  The proposed revision to DCD Section 9.2.2.4.5.2 provides an adequate description of the automatic isolation function. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

  The applicant proposed to modify the closure logic for CCS motor-operated containment isolation valves CCS-PL-V200, CCS-PL-V207, and CCS-PL-V208 to add a requirement to close on generation of the RCP bearing water high temperature pump trip signal. If a tube break occurs at normal (higher) RCS temperatures, the RTDs used for the automatic isolation function sense the temperature of the reactor coolant flowing through the collection header almost immediately after the break initiates. However, for an idled RCP with the RCS at low temperatures, such as near 93 °C (200 °F), RCS flow is much less (as low as 5 percent) than the flow rates seen at 100percent shaft speed. Even with reactor coolant temperature near 93 °C (200 °F), the RTDs will still sense the reactor coolant temperature within minutes and produce the automatic RCP bearing water high temperature trip, which closes the CCS containment isolation valves. Therefore, the staff’s review finds the applicant’s design change to using reactor coolant temperature instruments to initiate CCS containment isolation acceptable.

- The change to add Modes 3 and 4 to Table 3.3.2-1 of the TS is acceptable to the staff since it envelops the possible Modes within which this event is postulated to occur. The staff had additional questions concerning how the applicant would affect this trip in Modes 3 and 4 since the temperature differential between the RCPs and CCS may be so small that a trip signal would be delayed longer than necessary to effectively isolate the system. In its response, the applicant provided further details of the sensor arrangement and flow path of coolant for this event, should it occur in Modes 3 or 4. The staff found this discussion acceptable because it describes how the trip signal will be generated in sufficient time to isolate the system, without the need for permissive or interlocked trip signals. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- The applicant proposed to add a signal to Functional Diagram 7.2-1 (Sheet 5 of 20) in AP1000 DCD Tier 2 to close the component cooling system containment isolation valves. This closing signal is derived from the existing 2-out-of-4 control logic with bypass capacity of the high RCP bearing water temperature for any RCP. The high temperature setpoint and dynamic compensation for this new closing signal are the same as those used for RCP trips based on the high RCP bearing water temperature, but it will be implemented with a preset time delay. The newly added logic provides safety-related protection system functions and also meets the reliability and test requirements for the safety protection system as required by GDC 20 and GDC 21. Hence, the staff concludes that this change is acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

  The applicant has also proposed to remove the automatic control function to isolate the CCS RCP HX outlet isolation valves on a high delta-flow between the HX inlet and outlet flow. The high delta-flow between the HX inlet and outlet cooling water lines now will only generate a flow deviation alarm in the MCR. The original automatic isolation function is provided by the plant control system (PLS) and non-Class 1E sensors and instrumentation. This nonsafety-related isolation control function is redundant to and is replaced by the automatic closure of the CCS containment isolation valves based on the high RCP bearing water temperature produced through the safety-related PMS using safety-related RCP RTD sensors. In addition, a new alarm for the high delta-flow between the CCS RCP HX inlet and outlet cooling water lines is now provided in the MCR, and also the remote manual operation of CCS RCP HX outlet isolation valves is retained. If annunciated in the MCR, the new alarm would alert plant operators to close the RCP HX outlet isolation valve remotely in the MCR on the cooling water outlet line to prevent reactor coolant flow throughout the CCS. The staff finds that the above changes meet relevant criteria as required in 10 CFR 50.55a(h) and 10 CFR Part 50, Appendix A GDC 13 and GDC 19. The staff, therefore, concludes that these changes are acceptable.

23.V.4 Conclusion

The staff's review concludes that these proposed design changes are acceptable because they will not adversely affect safety-related SSCs; and the capability of the CCS to perform its defense-in-depth and RTNSS functions will not be degraded by the proposed changes. Adequate overpressure protection of the CCS is provided with ASME Code relief valves during a postulated RCP external heat exchanger tube rupture. An automatic signal will be generated to the associated containment isolation valves in the event of a postulated RCP external heat exchanger tube rupture, as shown in TS Table 3.3.2-1.

On the basis of its review of the containment isolation design aspect of the proposed CCS overpressure protection design change, the staff concludes that the design complies with the acceptance criteria in Section 6.2.4 of NUREG-0800, including: GDC 16; GDC 54; RG 1.26; and RG 1.29.

The staff also finds the design change to provide CCS overpressure protection by the two safety related relief valves meets the intention of the guidance for ISLOCA found in SECY-90-016.

23.W Changes to Add a Vacuum Relief System to the Containment

23.W.1 Description of Proposed Changes

In letters dated August 16, 2010; September 29, 2010; and October 15, 2010, the applicant submitted proposed design changes and supporting documentation, which add a containment vacuum relief system to the existing containment air filtration system (VFS) vent line penetration. The NRC staff reviewed the system design, containment external pressure analyses, containment isolation functions, leak rate testing, descriptions of the valve design, qualification and IST programs, I&C, and associated TS.

The applicant's proposed design changes add a vacuum relief system to the existing VFS 40.64-cm (16-in) vent line penetration as seen in DCD Tier 2, Figure 9.4.7-1, "Containment Air Filtration System P&ID", Sheet 1 of 2. The proposed vacuum relief system consists of redundant vacuum relief devices sized to prevent differential pressure between containment and the shield building from exceeding the design value. Each of the two vacuum relief device flow paths consists of a check valve (VFS-PL-V803A/B) inside containment; a motor operated butterfly valve (VFS-PL-V800A/B) outside containment; and associated piping. Each of these four valves also has a containment isolation function. The redundant check valves inside containment share a common inlet line with the vent line penetration and have independent discharge lines into containment. The redundant motor operated butterfly valves outside containment share a common inlet line. Each relief device, consisting of a check valve, connecting piping, and a motor operated valve (MOV), is designed to provide 100 percent of the required capacity to prevent a differential pressure across the containment vessel from exceeding the design value. Each relief flow path provides the required capacity, such that a single failure of any of the relief devices would not limit the flow below what is required to mitigate a containment vacuum relief event.

The normally closed butterfly valves are designed with motor operators that are powered from separate Class 1E direct current (dc) battery sources. They are designed to close within 30 seconds of receipt of either an automatic containment isolation signal, or a manual isolation signal. They are designed to open automatically within 30 seconds when the containment pressure signal reaches Low-2 level and remain open to preclude exceeding the containment external design pressure. While the vacuum relief system MOVs are open, the containment will be at a vacuum and flow will be into containment. Once the vacuum condition inside containment is reduced to near ambient pressure conditions, the open signal would automatically clear. This would allow the vacuum relief system MOVs, VFS-PL-V800A/B, to close automatically in the event that a containment isolation signal or high radiation signal is present. The check valves are balanced to remain closed during normal operations, including containment vessel venting.

The proposed vacuum relief system is being added to the existing VFS. A short description of the vacuum relief system is being added to DCD Section 9.4.7. Additionally, the proposed vacuum relief system design is shown on DCD Tier 2, Figure 9.4.7-1, Sheet 1 of 2, "Containment Air Filtration System P&ID." Since the proposed vacuum relief system valves are also part of the containment isolation design, valves VFS-PL-V800A/B and VFS-PL-V803A/B have been added to DCD Tier 2, Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves," with two notes. The first, Note 8, indicates that the valves VFS-PL-V800A/B close on either a containment isolation signal or a high radiation in containment signal, and open on a Low-2 containment pressure signal. The second, Note 9, indicates that the Type C leak rate testing of the valves VFS-PL-V800A/B would be in the reverse direction and that closure would occur in 30 seconds.

The applicant has changed the containment external design pressure specified in DCD Section 3.8.2.1.1 from 20.0 kPa (2.9 pounds per square inch differential (psid)) to 11.7 kPa (1.7 psid) based on the actuation point of the vacuum relief system. The applicant has also changed the containment external design analysis in DCD Section 6.2.1.1.4 to demonstrate the vacuum relief system is sufficient to mitigate the maximum expected external pressure.

The applicant has also modified AP1000 DCD Tier 2, Table 3.9-16, “Valve Inservice Test Requirements,” to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The IST table will specify VFS-PL-V800A/B as motor-operated butterfly valves with safety-related missions to maintain close, transfer close, maintain open, and transfer open; safety functions as active, containment isolation, safety seat leakage, and remote position; ASME Code Class 2 and IST Category A valves; and IST type and frequency as remote position indication and exercise every 2 years, containment isolation leak test, exercise full stroke quarterly and operability test. The IST table will specify check valves VFS-PL-V803A/B as relief valves with safety-related missions as maintain close, transfer close, and transfer open; safety functions as active, containment isolation, and safety seat leakage; ASME Code Class 2 and IST Category AC valves; and IST type and frequency as containment isolation leak test, exercise full stroke every refueling shutdown, and vacuum relief test every 2 years.

The applicant’s proposed design changes include the addition of TS 3.6.10, “Vacuum Relief Valves,” which address the proposed containment vacuum relief valve operation. The proposed design changes add a function to Table 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” for the signal which would open the containment vacuum relief valves on a Low-2 containment pressure level. In addition, the proposed changes extend the Mode Applicability for TS 3.6.4, “Containment Pressure,” and TS 3.6.5, “Containment Air Temperature,” to include Modes 5 and 6, which support TS 3.6.10 requirements.

The applicant has also proposed corresponding changes to Tier 1 DCD sections.

23.W.2 Regulatory Basis

The regulatory basis for evaluating the applicant’s proposed design changes and supporting documentation is documented in Chapters 3, 6, 7, 9, and 16 of NUREG-1793.

The applicable regulations for the design, analyses, containment isolation, and containment leak rate testing aspects of the applicant’s proposed design changes are detailed in NUREG-0800 Sections 6.2.1.1.A, “PWR Dry Containments”; 6.2.1, “Containment Functional Design”; 6.2.4, “Containment Isolation System”; 6.2.6, “Containment Leakage Testing”; and 9.4.3, “Auxiliary and Radwaste Area Ventilation System.” Specifically, the following regulatory requirements and guidance apply:

- 10 CFR Part 50, Appendix A, GDC 2
- 10 CFR Part 50, Appendix A, GDC 16
- 10 CFR Part 50, Appendix A, GDC 38

- 10 CFR Part 50, Appendix A, GDC 50
- 10 CFR Part 50, Appendix A, GDC 52
- 10 CFR Part 50, Appendix A, GDC 53, “Provisions for Containment Testing and Inspection”
- 10 CFR Part 50, Appendix A, GDC 54
- 10 CFR Part 50, Appendix A, GDC 56
- 10 CFR Part 50, Appendix J
- 10 CFR 50.34(f), “Contents of applications; technical information,” regarding Three Mile Island (TMI) Action Plan Item II.E4.2, “Containment Isolation Dependability”
- 10 CFR 52.47(b)(1), which requires that a DC application include the appropriate ITAAC
- RG 1.26
- RG 1.141, “Containment Isolation Provisions for Fluid Systems”
- BTP 6-4, “Containment Purging During Normal Plant Operations.”

The applicable regulations for the staff’s review of the functional design, qualification, and IST programs for the valves described in these proposed design changes are detailed in NUREG-0800 Section 3.9.6 (Revision 3), “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints.” The staff also considered guidance provided in applicable Commission SECY papers, Commission SRM, GLs, RGs, and regulatory issue summaries. Specifically, the following regulatory requirements apply:

- 10 CFR Part 50, Appendix A, GDC 1
- 10 CFR Part 50, Appendix A, GDC 2
- 10 CFR Part 50, Appendix A, GDC 4
- 10 CFR Part 50, Appendix A, GDC 54
- 10 CFR 50.55a(f), which requires that valves whose function is required for safety be assessed for operational readiness in accordance with the applicable revision to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code)
- 10 CFR 50.55a(b)(3), which takes exception to, or supplements, the ASME OM Code provisions for components within the scope of the IST Program

The applicable regulations for the I&C aspects of the applicant’s proposed design changes are detailed in NUREG-0800 Section 7.3, “Engineered Safety Features,” and NUREG-0800

Section 7.5, "Information Systems Important to Safety." Specifically, the following regulatory requirements and guidance apply:

- 10 CFR Part 50, Appendix A, GDC 13
- 10 CFR Part 50, Appendix A, GDC 19
- 10 CFR Part 50, Appendix A, GDC 20
- 10 CFR Part 50, Appendix A, GDC 21
- 10 CFR 50.55a(h), which requires that protection systems must meet the requirements stated in either IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995
- RG 1.97

The applicable regulations for evaluating the applicant's proposed TS are detailed in NUREG-0800 Chapter 16, "Technical Specifications – Revision 3, March 2010." The staff's evaluation focused primarily on confirming that changes to the GTS reflect the containment systems design and operating information in DCD Sections 7.3 and 9.4.7, and the containment analyses described in DCD Section 6.2, and that changes to the GTS meet the requirements of 10 CFR 50.36, "Technical specifications."

23.W.3 Evaluation

23.W.3.1 System Design and Analyses

The staff reviewed the proposed design changes for compliance with GDC 16, which requires that the reactor containment and associated systems be designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The proposed vacuum relief system includes an interlock with the inboard containment isolation valve for the containment purge line. The interlock prevents the vacuum relief system from opening if the inboard isolation valve is open. The staff requested that the applicant explain how the safety function would be accomplished if the inboard valve was open and the interlock prevented the vacuum relief system from functioning. The applicant responded that the vacuum relief system actuation had priority over the interlock and that if a containment vacuum condition existed, the system would actuate regardless of the interlock. The applicant revised the statements in the DCD to clarify this design feature. Specifically, the new DCD Section 7.6.2.4 states that if a vacuum relief actuation signal is present, the vacuum relief signal takes precedence over the valve closure interlock. This design feature ensures the safety function is accomplished and removes the possibility of a single failure associated with the interlock disabling the system.

The staff also reviewed the revised external pressure analysis in DCD Section 6.2.1 for compliance with GDC 16. Conformance with acceptance criteria from NUREG-0800 Section 6.2.1.1.A formed the basis for concluding whether GDC 16 was satisfied.

In DCD Revision 15, the applicant determined that the worst case event for maximum external pressure was the loss of all ac power sources during extreme cold weather. The analysis was conducted using the WGOTHIC code with the following conservatisms:

- External temperature boundary condition set at the minimum site parameter from DCD Tier 1, Table 5.0-1, -40 °C (-40 °F).
- Initial containment relative humidity set to 100 percent to maximize the vapor content and allow for the greatest reduction in pressure due to steam condensation.
- Loss of ac power event immediately reduces the containment heat load to zero.
- Initial containment temperature set at the maximum temperature LCO from TS 3.6.5, 48.8 °C (120 °F).
- Initial containment pressure set at the minimum pressure LCO from TS 3.6.4, -1.38 kPa (-0.2 psid).
- The velocity of air flowing over the containment shell is set at 7.56 meters per second (m/s) 24.8 feet per second (ft/s), which correlates to an external wind speed of 77.24 kilometers per hour (km/h) (48 miles per hour (mph)).
- No air leaks into containment.

The analysis demonstrated the differential pressure across the containment vessel remained below the design value for one hour, which was found to be sufficient time for plant operators to take action to mitigate the event.

The applicant revised the external pressure evaluation to incorporate the vacuum relief system and eliminate the need for operator action, to remove unnecessary conservatisms associated with the first three bulleted items above (no changes were made to the final four assumptions), and to incorporate design changes associated with the DCD amendment. The analysis was based on the WGOTHIC evaluation model approved by the staff in Section 23.S of this report, modified for use in an external pressure transient. The worst case event remains the loss of all ac power during cold weather. The applicant recognized that an operating reactor does not produce enough heat to raise the internal temperature to the maximum value on a cold day; thus, the combination of the maximum internal temperature and minimum external temperature is non-mechanistic. In order to remove conservatism, the first step in the revised analysis is determining the minimum external temperature that can sustain the internal temperature at 48.8 °C (120 °F). This pre-transient stage of the analysis assumed a heat rate equal to the value used to size the active containment cooling system. In order to minimize the external shell heat transfer coefficient, the annulus air flow was set to natural convection. The applicant used this model to determine equilibrium containment temperatures associated with various external temperatures and found that -3.88 °C (25 °F) is the minimum external temperature capable of maintaining the maximum internal temperature (48.8 °C (120 °F)).

The next step in the analysis is the loss of power transient. In this phase, the initial temperatures of the internal containment volume, containment shell, baffle and shield building were set to the equilibrium values found during the pre-transient run. For the design basis run, the internal temperature is 48.8 °C (120 °F) and the external temperature is -3.88 °C (25 °F), which bounds the proposed LCO for TS 3.6.10 requiring the internal/external temperature differential be less than or equal to 32.2 °C (90 °F). Because the initial conditions exceed the TS LCO, the equilibrium temperatures derived for the shell, shield building and baffle were also found to be bounding.

The original analysis assumed zero heat loss during the transient. This is non-mechanistic because it neglects the contribution of reactor system sensible and decay heat. In order to remove some of this conservatism, the heat load to containment is assumed equal to the sensible heat from a reactor in Mode 3 at normal operating temperature and normal operating pressure. Decay heat is not considered because it is assumed the reactor has never been critical. The staff agrees that these assumptions produce conservatively low heat loads. The applicant stated that the previous assumption of 100 percent relative humidity was also non-mechanistic because the temperature of the containment vessel is below the dew point. As a result, the revised analysis uses a relative humidity of 82 percent, representing a 25 percent margin over the equilibrium value. The staff finds this approach realistically bounds the relative humidity.

During the transient, the external temperature is assumed to decrease -1.11 °C (30 °F) per hour from an initial value of -3.88 °C (25 °F) until it reaches the minimum site value of -40 °C (-40 °F) at 7800 seconds. To demonstrate this assumption is bounding, The applicant evaluated hourly meteorological data gathered at Charlotte, North Carolina from January 1, 1975 to June 24, 1996 and at Duluth, Minnesota between January 1, 1975 and January 1, 2010. Charlotte, North Carolina was chosen by the applicant as a location having typical meteorological behavior for the Southeast Regional Climate Zone. The applicant also chose the Duluth, Minnesota location because it represents the basis for the AP1000 DCD minimum allowable operation temperature of -40 °C (-40 °F). The applicant found that the maximum observed hourly temperature decrease in Charlotte, North Carolina was -6.66 °C (20 °F) (from 22.77 °C (73 °F) to 11.66 °C (53 °F)). The applicant also found that the maximum observed hourly temperature decrease in Duluth, Minnesota, during below-freezing conditions was -8.33 °C (17 °F) (from -7.22 °C (19 °F) to -16.66 °C (2 °F)). The staff finds the applicant's use of a -1.11 °C (30 °F) per hour temperature drop to be reasonable because the applicant's analysis included several recent years worth of data at a typical southeast regional site (Charlotte, North Carolina) and a typical cold weather site (Duluth, Minnesota), and because there is significant margin between the observed values (-6.66 °C (20 °F) for Charlotte, North Carolina and -8.33 °C (17 °F) for Duluth, Minnesota) and the assumed hourly decrease (-1.11 °C (30 °F).

The WGOTHIC model incorporated one 15.2-cm (6-in) valve with a conservatively calculated system resistance, designed to open 20 seconds after the internal containment pressure reached the setpoint of -8.3 kPa (-1.2 psid). The safety analysis limit -8.3 kPa (-1.2 psid) is clearly identified in the DCD and will be used to develop the Low-2 containment pressure setpoint under the TS setpoint methodology program. The 20 seconds required to develop full flow is consistent with the mechanical design requirements identified in Section 9.1.1 of Enclosure #4 of the applicant's October 15, 2010 submittal. Because it is reasonable to expect a butterfly valve to allow significant flow at 60 percent of the stroke, this is also consistent with the proposed acceptance criteria for ITAAC Item 2 from Table 2.7.6-2 requiring the valves to open within 30 seconds.

The transient response demonstrated that the vacuum relief system limits the containment pressure to a minimum value of -11.2 kPa (-1.63 psid), which is bounded by the design value of

-11.7 kPa (-1.7 psid). As described in Enclosure #6 of its October 15, 2010, submittal, the applicant ran additional vacuum relief scenarios at external temperatures of -40 °C (-40 °F) and 10.0 °C (50 °F) and used the results to confirm that the design basis case was limiting. The -40 °C (-40 °F) case, initiated with an internal temperature of 31.11 °C (88 °F), was also used to demonstrate that the internal to external temperature differential LCO is not required if the internal containment temperature is less than 31.11 °C (88 °F). This bounds TS 3.6.10 Required Action B.2 to reduce containment average temperature to ≤ 26.66 °C (≤ 80 °F) when the inside to outside differential air temperature does not meet the LCO.

The staff reviewed the analysis basis during August 27, 2010, and October 3, 2010, audits of APP-SSAR-GSC-112, “AP1000 External Pressure Analysis to Confirm Sizing of the Vacuum Relief System,” Revision 0 and Revision 1. This report states that while various transients were considered for the external pressure evaluation, the loss of ac power event remains the most limiting. This is the same event that was found to be bounding in the original analysis, for the reasons discussed in Section 6.2.1 of NUREG-1793. The staff finds this evaluation remains applicable to the revised design because the proposed design changes will not impact determination of the limiting event.

The analysis applied a 7.56 m/s (24.8 ft/s) annulus velocity during the transient to represent an external wind of 77.24 km/h (48 mph). The staff finds this will produce conservatively high heat transfer coefficients because the basis for the correlation was testing on a prior version of the shield building. As discussed in Section 23.S, there is an increased pressure drop associated with the revised shield building, so the annulus velocity associated with a 77.24 km/h (48 mph) wind speed will be less than 7.56 m/s (24.8 ft/s).

The staff conducted an additional audit on August 30, 2010, on three supporting calculation notes. The staff found the baseline WGOTHIC external pressure model, described in APP-SSAR-GSC-746, Revision 0, acceptable because it used the same methodology as the DCD Revision 15 analysis to transform the LOCA model into an external pressure model. The staff reviewed the calculations for the total system resistance in APP-VFS-M3C-224, “Containment Vacuum Relief System Resistance Calculation,” Revision 0. The analysis was found to be conservative because the resulting value represents the resistance associated with the more limiting single flow path configuration (which assumes two failures) rather than the worst case single failure. It also incorporates a 65 percent margin over the calculated value to account for potential design changes. The staff reviewed the design basis for the minimum heat load into containment as described in APP-SSAR-GSC-003, “Calculation of the Total Loss of RCS Heat from Mode 3 with Station Blackout,” Revision 0. The staff finds this analysis to be conservative because the code used to generate the heat load does not consider contributions from the major component support structures, which are typically large contributors to containment sensible heat.

The staff performed a confirmatory analysis using the CONTAIN computer code and the AP1000 model described in Section 23.S.3 of this report. The model was altered to incorporate one vacuum relief line and to remove the shield building in order to apply the annulus air velocity directly to the external containment shell, producing conservatively high heat transfer coefficients on this surface. The model incorporated the same valve parameters as the applicant except the valve opening delay time was increased to 30 seconds. The staff evaluated the design basis case and the two sensitivity studies. The containment internal pressure transients demonstrated the design basis case, with a minimum negative pressure of -10.8 kPa (-1.56 psid), was limiting. These results are consistent with the applicant’s evaluation.

NUREG-0800 Section 6.2.1.1.A recommends that the margin between the external design pressure and the conservatively calculated minimum pressure be at least 10 percent at the construction permit (CP) or DC stage of review. The requirement at the operating license stage of review is that the maximum calculated value be less than the design value (zero-margin criterion). The more restrictive margin is applied at the CP and DC stage to account for revised or upgraded analytical models or minor changes that may result in a decrease in the margin in the as-built design. The margin in the revised design basis analysis is 4 percent. The applicant concluded that this margin is acceptable because the calculation of the vacuum relief system resistance included a 65 percent margin to accommodate system variances associated with plant construction. The applicant claims this addresses the "as-built" considerations that form the basis for NUREG-0800 recommended 10 percent margin at the DC stage. Furthermore, when the margin was removed from the loss coefficient, the resultant minimum external pressure was 10.1 kPa (1.46 psid), which provides a 14 percentmargin to the design value. The staff agrees this is an acceptable approach because it meets the intent of the NUREG-0800 recommendation. This finding is consistent with the staff evaluation in NUREG-1793 Section 6.2.1 where a zero-margin criterion was deemed acceptable for the AP1000 peak accident pressures at the DC stage.

The staff reviewed the proposed containment vacuum relief design for compliance with the requirements of GDC 2, which requires that the system be capable of withstanding the effects of earthquakes. The new portions of the system are safety-related and seismic Category I and, therefore, meet the requirements of GDC 2. However, in Section 9.4.7 of the DCD, the applicant removed statements that demonstrate the nonsafety-related portions of the VFS system meet GDC 2. The staff requested that the applicant explain how the VFS system meets GDC 2. The applicant revised the discussion in DCD Section 9.4.7 to state that system equipment and ductwork whose failure could affect the operability of the safety-related systems or components are designed to seismic Category II requirements. This conforms to the certified DCD Revision 15 design, and is adequate to demonstrate compliance with GDC 2.

As a result, the staff finds that the design and design features described in the DCD ensure the requirements of GDC 2 and GDC 16 are met. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue. The staff's evaluation of the AP1000 steel containment, including load combinations and design and analysis procedures, is described in Section 3.8.2 of this report.

23.W.3.2 Containment Isolation and Leak Rate Testing

The staff reviewed the proposed vacuum relief system design for compliance with GDC 56, which requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment be provided with containment isolation valves. One acceptable design includes: one automatic isolation valve inside, and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. The design of the vacuum relief system complies with the requirements of GDC 56.

10 CFR 50.34(f)(2)(xiv)(B) requires that each non-essential penetration (except instrument lines) shall have two isolation barriers in series. The vacuum relief design consists of two lines, which connect directly with the containment atmosphere and penetrate the primary containment. This design complies with the requirement of GDC 56 by providing each vacuum relief device with a check valve inside containment and a motor operated butterfly valve outside containment. This design complies with the 10 CFR 50.34(f)(2)(xiv)(B) redundancy requirement. If a check

valve failed to close during an accident, the MOVs in series with it would close on a “T” signal (described in DCD Section 6.2.3), thereby providing containment isolation.

To meet the requirements of GDC 56, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. All power-operated isolation valves should have position indications in the MCR. The safe post-accident position for the valves in the vacuum relief system is closed. The motor operated butterfly valves would close on a “T” containment isolation signal. To improve the reliability of the isolation function, these valves also close on a high radiation in containment signal. These valves are each powered by Class 1E batteries to ensure that they would close on these signals. Their position indication in the MCR is shown on DCD Table 6.2.3-1. Valves VFS-PL-V803A and VFS-PL-V803B are self actuated check valves, which would close if a vacuum does not exist inside containment. These check valves would either close or remain closed post-accident. Since the proposed vacuum relief system valves are also part of the containment isolation design, VFS-PL-V800A/B and VFS-PL-V803A/B have been added to DCD Tier 2, Table 6.2.3-1, “Containment Mechanical Penetrations and Isolation Valves,” with two notes. The first, Note 8, indicates that the valves VFS-PL-V800A/B close on either a containment isolation signal or a high radiation in containment signal, and open on a Low-2 containment pressure signal. The second, Note 9, indicates that the Type C leak rate testing of the valves VFS-PL-V800A/B would be in the reverse direction and that closure would occur in 30 seconds. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff also reviewed these proposed design changes for compliance with the requirements of GDC 54, as it relates to providing piping systems penetrating the containment with containment isolation capabilities having redundancy and reliability which reflect their importance to safety. To meet the reliability requirements, the components performing a containment isolation function are acceptable if Group B quality standards, as defined in RG 1.26, apply, and the components are designated seismic Category I in accordance with RG 1.29.

The containment isolation section of the vacuum relief design, consisting of the containment isolation valves and the included piping, are designed to ASME Code Section III, Class 2 criteria. The containment penetrations are classified as Quality Group B, as defined in RG 1.26, and seismic Category I. These designations are shown in DCD Tier 2, Figure 9.4.7-1 and Table 6.2.3-1, and Tier 1, Table 2.2.1-1, Table 2.2.1-2, and Figure 2.2.1-1, “Containment Isolation System.” The applicant has selected the appropriate mechanical design classification.

10 CFR 52.47(b)(1) requires that a DC application include the appropriate ITAAC. The ITAAC for the containment isolation MOV closing time are shown in DCD Tier 1, Table 2.7.6-2 and Table 2.2.1-3. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In meeting the requirements of GDC 54, relating to lines which provide open paths from the containment to the environs, such as containment purge and vent, the closure times of the isolation valves should minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. CSB BTP 6-4 provides additional guidance on the design and use of the containment purge systems, which may be used during the normal plant operating modes.

The VFS is used to purge the containment atmosphere of airborne radioactivity during normal plant operation. The proposed containment vacuum relief system is a safety grade system used to mitigate a containment external pressure scenario, and is part of the VFS, sharing the same containment penetration. The purge system is designed in accordance with CSB BTP 6-4. The purge component of the VFS uses 40.6-cm (16-in) supply and exhaust lines and containment isolation valves designed to close in 10 seconds. The vacuum relief component of the VFS uses 15.2-cm (6-in) supply lines and containment isolation valves designed to close in 30 seconds.

In the event of a LOCA, a maximum time of 30 seconds for closure of the 40.6-cm (16-in) valves was assumed for the analysis for the radiological consequences, as seen in DCD Tier 2, Table 15.6.5-2. This closure time is conservative, as the valve design closure time is 10 seconds. This closure time is consistent with the guidance in CSB BTP 6-4, and was found acceptable as an assumed closure time in the radiological analysis in NUREG-1793, Section 6.2.4.13. The 30 second closure times of the two new 15.2-cm (6-in) vacuum relief valves would be bounded by the current design, as the radiological consequences following a LOCA have already been found acceptable for two open 40.6-cm (16-in) valves, closing in 30 seconds.

To analyze the LOCA containment minimum backpressure, containment purge was assumed to be in operation at time zero and air was vented through both the 40.6-cm (16-in) diameter containment purge supply and exhaust lines until the isolation valves fully closed. These valves were modeled to close 12 seconds after the 55.2 kPa (8 psig) closure setpoint was reached, as described in DCD Tier 2, Section 6.2.1.5.3. This closure time was found acceptable as an assumed closure time in the ECCS analysis (reflood backpressure) in NUREG-1793, Section 6.2.4.13. In DCD, Tier 2, Section 9.4.7.2.1, the maximum time for closure of the vacuum relief valves, 30 seconds, was evaluated for its impact on the calculation of the LOCA backpressure. The minimum containment backpressure following two, 15.24-cm (6-in) valves closing in 30 seconds, is expected to be bounded by the containment backpressure resulting from two, 40.64-cm (16-in) valves closing in 12 seconds.

In NUREG-1793, Section 6.2.6, the staff reviewed the applicant’s proposed containment leakage rate testing program for AP1000 facilities described in DCD Tier 2, Section 6.2.5 and in the proposed TS of DCD Tier 2, Chapter 16. The staff reviewed the information in the DCD for conformance to 10 CFR Part 50, Appendix J, and to GDC 52 and GDC 53. GDC 52 requires the containment and associated equipment to be designed such that the periodic containment integrated leakage rate tests can be conducted at containment design pressure. GDC 53 requires that the containment allow periodic inspection, surveillance, and testing of certain SSCs.

The staff used the guidance, staff positions, and acceptance criteria of NUREG-0800 Section 6.2.6 and RG 1.163, “Performance-Based Containment Leak-Test Program,” in conducting its review.

The staff concluded that the AP1000 containment leakage rate testing program complied with the acceptance criteria of Section 6.2.6 of NUREG-0800 by satisfying the containment leakage rate testing requirements of GDC 52 and GDC 53, and Appendix J to 10 CFR Part 50.

The proposed vacuum relief system valves, VFS-PL-V800A/B and VFS-PL-V803A/B are also part of the containment isolation design and have been added to AP1000 DCD Tier 2, Table 6.2.3-1, “Containment Mechanical Penetrations and Isolation Valves.” These new containment isolation valves will be included in the AP1000 containment leak rate test program, and will be Type C tested, as indicated in DCD Tier 2, Table 6.2.3-1. Adding the new containment isolation valves to the already approved containment leak rate test program, meets the acceptance criteria in Section 6.2.6 of NUREG-0800 for containment leak rate testing of the new valves.

With regard to 10 CFR 52.47(b)(1), the applicant has added appropriate ITAAC to Tier 1. The applicant has included ITAAC to verify the valves function in both the open and closed directions. The staff finds the additional ITAAC acceptable.

23.W.3.3 Valve Design, Qualification, and Testing

The staff reviewed the functional design, qualification, and IST program descriptions for the valves to be used in the applicant's proposed containment vacuum relief system. The staff provided comments to the applicant related to the functional design, qualification, and IST program descriptions for the valves to be used in the proposed containment vacuum relief system. The applicant provided responses to all NRC requests for information. The staff also performed audits of the valve design specifications and data sheets referenced in the proposed design changes.

The proposed AP1000 containment vacuum relief system design includes parallel motor-operated butterfly valves VFS-PL-V800A/B outside containment connected by a single pipe line to parallel check valves VFS-PL-V803A/B inside containment. In the event of a vacuum condition within the AP1000 containment, butterfly valves VFS-PL-V800A/B will receive an automatic electric signal to their battery-powered motor actuators to open the valves. As a result, outside air will flow through the connected piping up to check valves VFS-PL-V803A/B that will open upon a preset differential air pressure across the valves and allow air to enter the AP1000 containment to relieve the vacuum condition. During its review of the applicant's submittal, the staff requested that the applicant clarify the design, qualification, and testing requirements for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The applicant provided this additional information in letters dated September 29, 2010, and October 15, 2010.

In Section 3.9.6 of this report, the staff concluded that the AP1000 DCD continues to provide sufficient information to satisfy 10 CFR Part 50 and 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants," for the design aspects of the functional design, qualification, and IST programs for safety-related valves to be used in the AP1000 reactor. The staff noted in Section 3.9.6 of this report that the operational program aspects regarding the functional design, qualification, and IST programs for safety-related valves will be reviewed as part of the evaluation of a COL application referencing the AP1000 certified design. The modifications to the AP1000 DCD related to the functional design, qualification, and IST programs for valves are addressed in Section 3.9.6.

In response to staff comments, the applicant revised Section 9.1.1, "Mechanical Design Requirements," in Section 9.1, "Outboard Motor Operated Valves VFS-PL-V800A/B," of its submittal to specify that the butterfly valves in the AP1000 containment vacuum relief system will be designed in accordance with AP1000 DCD Tier 2, Section 3.9, "Mechanical Systems and Components." The staff describes its review of the design requirements for safety-related MOVs in Section 3.9.6 of this report. For example, as indicated in the letter dated March 5, 2010, the applicant plans to revise Section 3.9 of the AP1000 DCD Tier 2 to specify that qualification of safety-related valves will be in accordance with ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants." The staff accepted the application of ASME Standard QME-1-2007 in Revision 3 to RG 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with certain staff positions. Also in response to staff comments, the applicant revised Section 9.1.1 of its submittal to specify the design requirements for capacity coefficient and stroke time for full flow capacity of these butterfly valves. The staff finds the reference to the provisions in AP1000 DCD Tier 2, Section 3.9, with the valve-specific design requirements in Section 9.1.1 of the applicant's submittal, to be acceptable for the functional design and qualification of the butterfly valves to be used in the proposed containment vacuum relief system with respect to the design aspects of the AP1000 DC amendment as discussed in Section 3.9.6 of this report.

During its review, the staff requested that the applicant describe the availability of adequate power supplies for motor-operated butterfly valves VFS-PL-V800A/B to perform their design-basis functions. Section 9.1.2, "Valve Electrical Requirements," of the applicant's submittal specifies the design requirements for butterfly valves VFS-PL-V800A/B to be powered from separate Class 1E dc battery sources. Section 9.1.2 also specifies that butterfly valves VFS-PL-V800A/B will be designed to stroke twice for their design-basis operation. Section 9.1.2 indicates that the electrical calculations using the methodology described in IEEE Standard 485, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," will take into consideration the starting current and stroke time for the MOV operations. Section 8.2, "DC Power Systems," in AP1000 DCD Tier 2 describes the application of IEEE-485 for dc power systems in the AP1000 certified design. In addition, the functional qualification of butterfly valves VFS-PL-V800A/B in accordance with ASME Standard QME-1-2007 as accepted in Revision 3 to RG 1.100 will demonstrate the power capability of these MOVs to perform their design-basis functions. The staff finds that the applicant has described the methodology for providing adequate power availability to butterfly valves VFS-PL-V800A/B in an acceptable manner.

In response to staff comments, the applicant revised Section 9.1.3, "Testing Requirements," of its submittal to specify that butterfly valves VFS-PL-V800A/B will be tested in accordance with AP1000 DCD Tier 2, Section 3.9.6, "Inservice Testing of Pumps and Valves." As discussed above, the staff describes its review of the description of the IST program for safety-related valves in Section 3.9.6 of this report. Therefore, the staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9.6 for the description of the IST program for the butterfly valves to be used in the AP1000 containment vacuum relief system to be acceptable with respect to the design aspects of the AP1000 DC amendment as discussed in Section 3.9.6 of this report. The staff will review the operational aspects for the IST program for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

Section 9.2.1, "Mechanical Design Requirements," of the applicant's submittal specifies the design requirements for the AP1000 vacuum relief containment isolation check valves. ASME OM Code, Table ISTC-3500-1 states that if a check valve used for pressure relief is capacity certified, then it shall be classified as a pressure or vacuum relief valve. Therefore, check valves VFS-PL-V803A/B are within the scope of ASME Code, Section III, Subsection NC-7000, "Overpressure Protection," as defined in Paragraph NC-7110, and require capacity certification as defined in Paragraph NC-7750. In response to staff comments, The applicant revised Section 9.2.2, "Testing Requirements," of its submittal to specify that VFS-PL-V803A/B are vacuum relief valves that will be designed and qualified in accordance with ASME Code, Section III, Subsection NC-7000. During its review, the staff informed the applicant that the design requirements for check valves VFS-PL-V803A/B should be included in Section 9.2.1 (rather than Section 9.2.2). As a result, the applicant modified Section 9.2.1 to specify:

(1) VFS-PL-V803A/B are vacuum relief valves that will be designed and qualified in accordance with ASME Code, Section III, Subsection NC-7000; (2) the valves will be qualified using the provisions in AP1000 DCD Tier 2, Section 3.9; (3) the valves will be designed with an allowable tolerance for the 1.38 kPa (0.2 psi) differential opening pressure; and (4) the valve flow capacity to relieve vacuum conditions to avoid the containment external design pressure from being exceeded. The staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9 and the ASME Code design requirements, with the valve-specific design requirements specified in Section 9.2.1 of the applicant's submittal, to be acceptable for the functional design and qualification of the check valves to be used in the AP1000 containment vacuum relief system.

Section 9.2.2 of the applicant's submittal specifies the testing requirements for AP1000 vacuum relief containment isolation check valves VFS-PL-V803A/B. Because these check valves provide a vacuum relief function with a design opening pressure, the NRC informed the applicant that its submittal should specify that these valves will be tested in accordance with ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants." In response, the applicant revised Section 9.2.2 to specify that these check valves are tested in accordance with AP1000 DCD Tier 2, Section 3.9.6 and ASME OM Code, Appendix I. Section 9.2.2 also states that these check valves will be tested in both directions in light of their safety functions in both the open and close directions. Section 9.2.2 specifies that 1.38kPa (0.2 psid) will be used as an acceptance criterion for the check valve opening test. As discussed above, the staff describes its review of the description of the IST program for safety-related valves in Section 3.9.6 of this report. Therefore, the staff finds the reference in the applicant's submittal to the provisions in AP1000 DCD Tier 2, Section 3.9.6 and ASME OM Code, Appendix I, for the description of the IST program for the check valves to be used in the AP1000 containment vacuum relief system to be acceptable with respect to the design aspects of the AP1000 DC amendment as discussed in Section 3.9.6 of this report. The staff will review the operational aspects regarding the IST program for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

In response to staff comments, the applicant modified Section 6.0, "Containment Isolation Consideration," of its submittal in the "Position" paragraph for 10 CFR Part 50, Appendix A, GDC 54, to clarify that butterfly valves VFS-PL-V800A/B cannot be tested in the direction of containment leakage. Therefore, these butterfly valves will be tested in the reverse direction of containment leakage. This testing is more conservative because the valves will be installed such that the containment pressure will assist in sealing the valve closed to minimize containment leakage. The applicant's submittal indicates the application of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8, "Containment System Leakage Testing Requirements," for containment valve testing as referenced in the AP1000 DCD. The staff requested that the applicant clarify that the butterfly valves will be installed in an orientation that provides for containment pressure to help seal the valve closed. In its response, the applicant revised Section 6.0 of its submittal to specify that the butterfly valves in the containment vacuum relief system will be installed such that containment pressure will assist in sealing the valve closed. The staff finds that the applicant has clarified the orientation of the butterfly valves in the containment vacuum relief system to support the direction of leakage testing for these valves.

As part of its submittal, the applicant plans to modify AP1000 DCD Tier 2, Table 3.9-16, "Valve Inservice Test Requirements," to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. The IST table will specify VFS-PL-V800A/B as motor-operated butterfly valves with safety-related missions to maintain close, transfer close, maintain open, and transfer open; safety functions as active, containment isolation, safety seat leakage, and remote position; ASME Code Class 2 and IST Category A valves; and IST type and frequency as remote position indication and exercise every 2 years, containment isolation leak test, exercise full stroke quarterly and operability test. The IST table will specify check valves VFS-PL-V803A/B as relief valves with safety-related missions as maintain close, transfer close, and transfer open; safety functions as active, containment isolation, and safety seat leakage; ASME Code Class 2 and IST Category AC valves; and IST type and frequency as containment isolation leak test, exercise full stroke every refueling shutdown, and vacuum relief test every 2 years. During its review, the staff requested that the applicant resolve an apparent inconsistency between the IST table and its Note 39. In a letter dated October 15, 2010, the applicant provided a planned revision to the IST table to clarify that the containment vacuum relief butterfly valves will be exercised quarterly and that the containment vacuum relief check valves will be full stroke exercised every refueling shutdown. The applicant’s submittal indicates that Note 39 to the IST table will justify the extended test interval for the check valves because of their location inside containment. The staff finds the planned provisions for IST activities for the butterfly valves and check valves in the containment vacuum relief system to comply with the requirements in 10 CFR 50.55a and the ASME OM Code and, therefore, to be acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff requested that the applicant clarify the provisions for inspection and maintenance of the valves in the AP1000 containment vacuum relief system. In its response, the applicant clarified that the inspection and maintenance of butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B are the same as those for other safety-related valves. These valves are required to satisfy the design, inspection, and testing requirements described in the applicable section of the AP1000 DCD. These valves will be designed for the full lifetime of their service, and will not require inspection following individual actuations. Corrective action will be taken for any identified malfunction. Periodic testing and inspection results will be documented and trended. The plant predictive maintenance program and its associated condition monitoring will include these valves. The staff finds that the applicant has clarified the implementation of plant programs for inspection and maintenance for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B that complies with the NRC regulations and, therefore, is acceptable.

The applicant’s submittal indicates that the scope of AP1000 DCD Tier 1, Section 2.2.1, “Containment System,” will be expanded to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. Table 2.2.1-3, “Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC],” for the containment system specifies ITAAC for piping and component design in accordance with ASME Code Section III; piping, components, and welds satisfying ASME Code, Section III requirements for integrity; seismic design-basis capability; Class 1E environmental qualification; valve operating times; and valve functional qualification. In its letter dated October 15, 2010, the applicant provided a planned change to Table 2.2.1-3 to specify the closing time for the butterfly valves in the AP1000 containment vacuum relief system. The applicant also indicated that Table 2.7.6-2 in AP1000 DCD Tier 1, Section 2.7.6, “Containment Air Filtration System,” will include new ITAAC for the opening time for the butterfly valves in the AP1000 containment vacuum relief system. The staff finds these ITAAC to be acceptable for the butterfly valves and check valves in the AP1000 containment vacuum relief system. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 9.3, "Valve Design Specifications and Datasheets," of the applicant's submittal indicates that the design specifications and data sheets for butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B were available for review at the Westinghouse office. On September 16, 2010, the staff reviewed the referenced documents at the Westinghouse office. During this audit, the staff found that the referenced valve specifications and data sheets had not been updated to include butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B. As a result, the applicant revised Section 9.3 in its submittal to describe the data sheets to be prepared for the butterfly valves and check valves to be used in the AP1000 containment vacuum relief system to support the applicable valve design specifications. On October 7, 2010, the staff audited the preliminary valve specifications and data sheets at the Westinghouse office. The staff finds that the applicant has provided an acceptable description of the valve design specifications and data sheets to support the design and qualification of the butterfly valves and check valves to be used in the AP1000 containment vacuum relief system.

ASME Code, Section III, Article NE-7000, "Overpressure Protection," Subsection NE-7150, "Acceptable Pressure Relief Devices," specifies in Article NE-7152 that vacuum relief devices shall meet the construction requirements applicable to Class 2 valves. The ASME Code also states that valve devices intended to provide vacuum relief, and which are operated by indirect means depending upon an external energy source, are not acceptable unless the following conditions are met: (1) at least two independent external power-operated valve and control systems are employed so that the required relieving capacity is obtained if any one of the valve systems should fail to operate when called upon to do so; and (2) at least one self-actuating vacuum relief device of equivalent relieving capacity is provided in series with each of the external power-operated valves. The ASME Code indicates that acceptable self-actuating vacuum relief devices include balanced self-actuating, horizontally installed, swing disk valves, with provisions for adjustment for the differential pressure under which the valves will operate.

For the proposed AP1000 containment vacuum relief system, Sections 9.1.1 and 9.2.1 of the applicant's submittal specify that butterfly valves VFS-PL-V800A/B and check valves VFS-PL-V803A/B will be designed as ASME Code, Class 2 valves. Motor-operated butterfly valves VFS-PL-V800A/B will receive an automatic electric signal to their battery-powered motor actuators to open the valves from separate Class 1E dc battery sources. Check valves VFS-PL-V803A/B will be designed as swing check valves installed in the horizontal direction with a 1.38 kPa (0.2 psi) differential opening pressure. The design of the proposed AP1000 containment vacuum relief system includes parallel check valves VFS-PL-V803A/B in series with the parallel butterfly valves VFS-PL-V800A/B. The staff finds that the design of the vacuum relief devices for the proposed AP1000 containment vacuum relief system satisfies the provisions in Subsection NE-7150 of ASME Code, Section III.

23.W.3.4 Instrumentation and Control

As documented in Chapter 7 of NUREG-1793, the staff reviewed and approved the engineered safety features and safety-related display information as specified in Sections 7.3 and 7.5, respectively, of the AP1000 DCD, Revision 15. The staff reviewed these proposed design changes using the review procedures described in NUREG-0800 Sections 7.3 and 7.5.

The applicant created a new functional diagram Figure 7.2-1 (Sheet 19 of 21) for controlling the proposed containment vacuum relief MOVs from the PMS. The proposed vacuum relief system MOVs are designed to close within 30 seconds by using the existing VFS isolation signal from the PMS, which includes closure from an automatic containment isolation signal, a High-1 containment radiation signal, a manual containment isolation, or a manual containment cooling signal. In accordance with these proposed design changes, the normally closed vacuum relief system MOVs would open automatically during either of the following conditions:

- 2-out-of-4 coincidence logic for a Low-2 containment pressure condition
- manual actuation of either of the two momentary controls

The 2-out-of-4 logic also receives signals that indicate whether divisions have been bypassed. If one division is bypassed for test or maintenance, the logic will automatically be modified to 2-out-of-3 coincidence logic. Actuating either of the two momentary manual controls in the MCR would actuate all applicable divisions. Separate momentary controls are also provided in the MCR for manually resetting the containment vacuum relief system. In these proposed design changes, the open control function has priority over the closing control function for the containment vacuum relief MOVs. In addition, in order to prevent the purge line isolation inside valve from being opened simultaneously with the vacuum relief system MOVs, interlock logic is implemented to make sure that the vacuum relief system MOVs could not be opened unless the purge line isolation inside valve is closed; if open, vacuum relief system MOVs would close.

While the vacuum relief system MOVs are open, the air flow would be into the containment, which is at a vacuum. Once the vacuum condition inside containment is reduced to near ambient pressure conditions, the open signal would automatically clear. This would allow the vacuum relief system MOVs to close automatically in the event that a containment isolation signal or high radiation signal is present.

The applicant included the necessary design change information in the revised Tier 2, Section 7.3.1, Tables 7.3-1 and 7.3-3, new functional diagram Figure 7.2-1 (Sheet 19 of 21), and Tier 1, Table 2.5.2-3 in the DCD. The applicant also added a new Section 7.6.2.4 in Tier 2 to address the interlock for the containment vacuum relief isolation system, which was added to Tier 1, Table 2.5.2-7 as well for the PMS Interlocks. Since the above logic included for the proposed containment vacuum relief system is designed with division redundancy and bypass capability, the staff concludes that the proposed containment vacuum relief system meets the applicable criteria of protection, reliability, and testability as required in 10 CFR 50.55a(h), and GDC 20 and GDC 21. The applicant added the status of the containment vacuum relief MOVs to Table 7.5-1 for post-accident monitoring, and revised Tier 1, Tables 2.5.2-3 and 2.5.2-4 to add the automatic and manual control functions in the PMS for the containment vacuum relief MOVs. The applicant revised Tier 1, Table 2.5.2-5 and Tier 2, Table 18.12.2-1 to include the display and control of the containment vacuum relief MOVs as part of the minimum inventory for the AP1000 DCD. On the basis of its review of the above changes, the staff finds that the design of the new containment vacuum relief system proposed by the applicant meets the applicable instrumentation and control requirements as mandated in GDC 13 and GDC 19, and also complies with the regulatory guidance in RG 1.97.

The staff finds that the proposed instrumentation and controls design changes meet the relevant requirements in 10 CFR 50.55a(h); 10 CFR 52.47; and 10 CFR Part 50, Appendix A, GDC 13; GDC 19; GDC 20; and GDC 21. The proposed design changes also conform to the related guidance in RG 1.97. Therefore, the staff concludes that the instrumentation and controls aspects of the proposed containment vacuum relief system are acceptable.

23.W.3.5 Technical Specifications

With respect to the proposed TS requirements in TS 3.3.2 and TS 3.6.10, and the proposed changes to TS 3.6.4 and 3.6.5, the staff finds these additions and changes acceptable because they conform to guidance in the Westinghouse STS (NUREG-1431, “Standard Technical Specifications — Westinghouse Plants”), and reflect the containment systems design information in DCD Sections 7.3 and 9.4.7, and their associated analyses described in DCD Section 6.2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

23.W.4 Conclusion

On the basis of its review and confirmatory calculations, the staff finds that the external pressure analysis meets the acceptance criteria from NUREG-0800 Section 6.2.1.1.A, and thus satisfies GDC 16.

On the basis of its review of the containment isolation design aspect of the proposed vacuum relief design, the staff concludes that the design complies with the acceptance criteria in Section 6.2.4 of NUREG-0800, including 10 CFR 50.34(f)(2)(xiv), and CSB BTP 6-4.

On the basis of its review, the staff concludes that the proposed addition of the vacuum relief valves to the already certified AP1000 containment leakage rate testing program complies with the acceptance criteria of Section 6.2.6 of NUREG-0800. Compliance with NUREG-0800 acceptance criteria provides adequate assurance that containment leaktight integrity can be verified before initial operation and periodically throughout its service life. Compliance with the criteria in Section 6.2.6 of NUREG-0800 constitutes an acceptable basis for satisfying the containment leakage rate testing requirements of GDC 52, GDC 53, and Appendix J to 10 CFR Part 50.

The staff reviewed the functional design, qualification, and IST program description for the valves to be used in the proposed AP1000 containment vacuum relief system in accordance with the regulatory basis described in this safety evaluation. Based on its review, the staff concludes that the applicant has provided an acceptable description of the design aspects of the functional design, qualification, and IST programs for the valves to be used in the AP1000 containment vacuum relief system. The staff will review the operational aspects regarding the functional design, qualification, and IST programs for safety-related valves as part of the evaluation of a COL application referencing the AP1000 certified design.

The staff finds that the proposed instrumentation and controls design changes meet the relevant requirements in 10 CFR 50.55a(h), and 10 CFR Part 50, Appendix A, GDC 13, GDC 19, GDC 20, and GDC 21.

On the basis of its review, the staff finds that the proposed changes to the TS that have been established for the containment systems meet the requirements of 10 CFR 50.36.

In addition to the specific DCD changes discussed above in this SER section, the applicant proposed additional changes to several sections, tables, and figures in AP1000 DCD Tier 2 Sections 3.2, 3.7, 3.8, 3.9, 3.11, 7.2, 7.7, 18.12, 19.55, 19.59, and Appendices 1A, 3I and 9A to incorporate the design, testing, and operation of the new containment vacuum relief system and its individual components. The staff has reviewed those proposed DCD changes and finds them to reflect the design, testing, and operation of the new containment vacuum relief system in an acceptable manner.

23.X Changes to the Passive Containment Cooling System

23.X.1 Description of Proposed Changes

The applicant has proposed changes to modeling of the Passive Containment Cooling System (PCS) to correct an error in the time for PCS to begin steady state film coverage of the containment vessel shell which is determined using a scaling factor from AP600 full scale 1/8 sector testing. The applicant assessed the impact of these changes to the PCS in APP-GW-GLR-096, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis," Revision 2, and submitted the report to the NRC in a letter dated March 10, 2011.

This report provides a scaling calculation of the time for PCS to begin steady state film coverage, a discussion of how these changes impact design basis test results, and results of evaluations of the limiting design basis accidents (DBA). Appendix B of APP-GW-GLR-096 contains proposed DCD changes. The changes that impact Section 6.2.1, "Containment Functional Design" of NUREG-1793 are as follows:

- Corrected value of the time for PCS to begin steady state film coverage.
- Update containment response to reflect results of analyses that use the Corrected value of the time for PCS to begin steady state film coverage.

The proposed changes also include revisions to technical specifications (TS). The applicant proposed to change the calculated peak containment internal pressure for the design basis loss of coolant accident to the updated value in TS Bases B 3.6.4, "Containment Pressure," and TS 5.5.8, "Containment Leakage Rate Testing Program".

23.X.2 Regulatory Basis

The following Commission regulations are related to the evaluation of the PCS:

- GDC 38, "Containment Heat Removal," as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels
- GDC 50, "Containment Design Basis," as it relates to demonstrating sufficient margin in accident analysis
- 10 CFR 52.47(c)(2) and 10 CFR 50.43(e), as they relates to design certification analysis and testing in support of a passive plant design.

The regulatory basis for evaluating the generic technical specifications (GTS) is documented in Chapter 16 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS design and operating information described in DCD Section 6.2. The proposed changes were evaluated using the guidance provided by Standard Review Plan (SRP) Chapter 16, "Technical Specifications – Revision 3, March 2010." Acceptability was based on conformance with the guidance in by SRP Chapter 16.

23.X.3 Evaluation

As described in APP-GW-GLR-096, Revision 2, the applicant estimated the total time to establish steady state film coverage for the AP1000 containment vessel is estimated to be 400 seconds. The applicant modified the WGOTHIC evaluation model that was used in previous DCD analyses to reflect the updated value of time to establish steady state film coverage. The applicant ran the design basis loss-of-coolant accident (LOCA) and main steam line break (MSLB) event cases using this revised model to determine the impact of the delay in PCS water application on the calculated pressure and temperature responses.

In its March 10, 2011, letter, the applicant states that in addition to the PCS water application delay, containment modeling changes included adding an epoxy top coat on the containment vessel to a wainscot height of 7 feet above the operating floor. The applicant states in the same letter that adding an epoxy top coat was to make the containment model and containment pressure analysis consistent with Section 6.1 of the DCD, which states the design requirement for the epoxy top coat. Applicant's sensitivity calculation has shown that the impact on containment peak pressure from adding an epoxy top coat the containment model was 0.01 psi.

Appendix B to APP-GW-GLR-096, Revision 2, provides DCD markup on tables and figures with calculated results for double-ended cold leg guillotine break LOCA, and full main steam line double-ended rupture (DER) at 30 percent and 101 percent reactor power with failure of main steam isolation valve. The results show that the containment pressure and temperature after a LOCA or MSLB would stay below the containment design pressure and temperature.

Figures 8-7 and 8-8 of APP-GW-GLR-096, Revision 2, show comparisons for a double-ended hot leg guillotine (DEHLG) LOCA containment response. As shown in the Figures 8-7 and 8-8, the increased delay time for the application for PCS flow has little or no impact on the calculated peak containment pressure and temperature for this short-term event because the transient pressure peak occurs before PCS flow is initiated.

During an audit performed at the Westinghouse Office in Rockville, Maryland between February 22 and March 4, 2011, the staff reviewed the applicant's calculation report supporting the results of analyses presented in APP-GW-GLR-096, Revision 2.

Based on its review of APP-GW-GLR-096, Revision 2, the staff determined that after correcting the delay in PCS water application in the containment analysis, the calculated pressure and temperature are below the containment design pressure and temperature, and therefore, acceptable. The audit the staff performed confirmed these conclusions.

The staff determined that the change in the delay in PCS water application in the containment analysis did not affect the staff's conclusions as stated in NUREG-1793 and the design is compliant with GDC 38, GDC 50, 10 CFR 52.47(c)(2), and 10 CFR 50.43(e).

During its review, the staff found the applicant's proposed changes to TS Bases B 3.6.4 and TS 5.5.8 acceptable because they reflect the peak containment pressure following a design-basis accident as given in DCD Tier 2 Section 6.2.

In AP1000 DCD Revision 19, the applicant made changes to the text, which are consistent with the proposed changes in the March 10, 2011 letter.

23.X.4 Conclusion

Based on its review the staff concludes that the design changes described above are acceptable and the design is compliant with GDC 38, GDC 50, 10 CFR 52.47(c)(2), and 10 CFR 50.43(e).

23.Y Changes to WGOTHIC AP1000 Containment Evaluation Model Inputs

23.Y.1 Description of Proposed Changes

The applicant has proposed changes to the WGOTHIC AP1000 containment evaluation model (CEM) inputs to address errors and to make updates. The applicant assessed the impact of these changes to the containment response to DBAs in APP-GW-GLR-096, Revision 3, and submitted the report to the NRC in a letter, dated June 14, 2011. The changes include the following, as stated in Section 7.2 of the report:

- The recalculation of the LOCA mass and energy (M&E) releases

- The removal of the inorganic zinc coatings within the LOCA zone of influence and within the maximum flood elevation inside the containment

- The revision of the specific heat for the inorganic zinc and epoxy coatings

- The revision of the material properties for the containment shell

- The correction of the modeling of heat transfer from the containment vessel shell below the operating deck

- The credit for some existing thermal conductors for platforms/gratings

- The consideration of the accumulator nitrogen gas after injection

Changes to the calculation of LOCA M&E releases are identified in Section 7.3 of APP-GW-GLR-096. They include:

- An increase in the SG pressure at the tube bundle
- An increase in the vessel metal mass
- A reduction in the SG tube heat transfer area
- A reduction in core power
- An increase in the RCS fluid volume
- An increase in the main feedwater flow
- A reduction in the equilibration temperature

Appendix B of APP-GW-GLR-096 contains proposed DCD changes. The changes to the WGOTHIC AP1000 CEM input resulted in a small increase of containment peak containment pressure, which impacts a number of sections of the DCD and Section 6.2.1, “Containment Functional Design” of NUREG-1793. Crediting some existing thermal conductors for platforms/gratings, which were not credited in the CEM before, also impacts Section 21, "Testing and Computer Code Evaluation" of NUREG-1793.

The proposed changes also include revisions to TS. The applicant proposed to change the calculated peak containment internal pressure for the design basis LOCA to the updated value in TS 5.5.8, "Containment Leakage Rate Testing Program," and TS Bases B 3.6.4, "Containment Pressure."

23.Y.2 Regulatory Basis

The Commission regulation related to the evaluation of the proposed changes to the CEM is GDC 50, "Containment Design Basis." GDC 50 requires the demonstration of sufficient margin in the containment for design basis accident analysis. The staff evaluated the proposed changes using the guidance provided in NUREG-0800 Sections 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments"; and 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)." Acceptability was based on conformance with the guidance in NUREG-0800 Sections 6.2.1.1.A and 6.2.1.3.

The regulatory basis for evaluating the GTS is documented in Chapter 16 of NUREG-1793. The staff's evaluation of the proposed changes focused primarily on confirming that the changes to the GTS reflect the PCS design and operating information described in DCD Section 6.2. The staff evaluated the proposed changes using the guidance provided by NUREG-0800 Chapter 16, "Technical Specifications." Acceptability was based on conformance with the guidance in NUREG-0800 Chapter 16.

23.Y.3 Evaluation

As described in APP-GW-GLR-096, Revision 3, the applicant modified the WGOTHIC CEM inputs that were used in previous DCD analyses to reflect the corrected and updated values. The applicant ran the design basis LOCA and MSLB event cases using this revised model to determine the impact of the input changes on the calculated pressure and temperature responses.

23.Y.3.1 **LOCA Mass and Energy (M&E) Releases**

The staff reviewed the long-term LOCA M&E release calculations and the methodology as it applies to the AP1000. The long-term LOCA M&E release calculation provides a key input for the containment evaluation model. The blowdown phase M&E releases were calculated using the SATAN-VI computer code, as described in WCAP-10325-P-A (proprietary) and WCAP-10326-A (nonproprietary), May 1983). The staff has previously determined that the SATAN-VI LOCA blowdown computer program is acceptable for use in obtaining LOCA M&E releases for the LOCA blowdown phase for containment analyses. SATAN-VI has been approved by the staff for this purpose, as discussed in NUREG-0800 Section 6.2.1.4. Applicability to the AP1000 is addressed in Section 6.2.1.3 of NUREG-1793.

The LOCA M&E release input data is documented in Tables 6.2.1.3-9 and 6.2.1.3-10 of the DCD. As described in Section 7.3 of APP-GW-GLR-096, a number of input values were corrected in the blowdown portion of the LOCA M&E release calculation, along with one input value in the long-term release spreadsheet calculation. These input changes produced slightly higher LOCA M&E releases to the containment.

The seven specific SATAN model changes made to the DCD, Revision 18 analysis were reviewed by the staff and audits were performed at the Westinghouse office in Rockville, Maryland between May 23 and June 24, 2011, to determine that the changes were appropriate. Each change is described in detail below.

a. *Steam Generator Pressure at the Tube Bundle*

The SATAN-VI model requires an input value for the SG secondary side pressure at the tube bundle. However, the input value that was used to calculate the M&E releases that are documented in DCD Revision 18 was based on the SG secondary side outlet pressure. The value used previously does not maximize the initial stored energy in the SGs secondary side. Correction of this value resulted in a higher calculated containment pressure response. The staff confirmed the code input requirements and the referenced design specifications for SG pressure, and agrees with the applicant that this change is appropriate.

b. *Vessel Metal Mass*

In the M&E calculation for DCD Revision 18, some of the metal mass associated with the reactor vessel core barrel region was not included in the SATAN-VI model. The total mass of the missing metal was approximately five percent of the total reactor vessel metal mass. The lower original input did not maximize the initial stored energy in the RCS metal. Correction of this value resulted in a higher calculated containment pressure response. The staff confirmed the revised input to the SATAN-VI M&E calculation.

c. *Steam Generator Tube Heat Transfer Area*

The tube heat transfer area input value for each SG in the M&E calculation for DCD Revision 18 was approximately seven percent larger than the correct value. Correction of the tube heat transfer area reduces the heat transfer between the RCS and SGs during the blowdown phase, but does not impact the SG energy release rate that is assumed in the long-term spreadsheet calculation. The staff confirmed the revised input to the SATAN-VI M&E calculation and agrees with the applicant that the long-term release rate assumed is unaffected by the change in input area.

d. *Core Power*

The power level input for the LOCA M&E release calculation was changed from the DCD Revision 18 analysis value of 3415 MWt to 3400 MWt. The original value included 15 MWt of RCP energy that is generated when all four pumps are energized. The RCPs are tripped on a LOCA. Additionally, the analysis methodology for a design basis large break LOCA assumes a loss of offsite power coincident with LOCA initiation. With a loss of offsite power, the RCPs would trip and begin coasting down. The SATAN code input for power is the licensed core power plus the plant-specific power uncertainty, which is included for conservatism. The input is used to initialize the reactor decay heat calculation. Including the energy that is equivalent to the energy generated by the RCPs when electrically powered is not necessary for the SATAN input because RCP energy does not contribute to the decay heat from the fuel. The revised core power is the licensed core power. The implemented change does not alter the modeling characteristics of the RCPs from the prior LOCA M&E release analysis or the approved methods in WCAP-10325-P-A. Changing this core power input value results in a slight reduction in the decay heat contribution to the event. With a loss of power, the RCPs will not continue to operate. However, the staff was concerned about the treatment in the analysis of the additional heat contributed by the RCPs upon coastdown. In an audit discussion and subsequently in a letter dated July 7, 2011, the applicant provided further justification for excluding the pump energy from the initial core power. The pump coastdown energy was shown by the applicant to be an extremely small contribution to the total energy release during a LOCA. Independent calculation using design data provided by the applicant confirmed the pump coastdown energy addition and duration. As a result, the staff finds the core power change in SATAN acceptable.

e. Reactor Coolant System Fluid Volume

The LOCA M&E release methodology uses fluid volumes that have been determined from engineering drawings. This is referred to as a “cold” condition, so the fluid volumes that are used as input to the SATAN-VI model were increased to account for thermal expansion and measurement uncertainty. The total fluid volume of the RCS in the original calculation for DCD Revision 18 was slightly underestimated. The applicant has corrected the values in the new M&E calculations and as such the staff finds the change acceptable. Although the change is small and the overall impact is insignificant the correct values are now used.

f. Feedwater Flow Rate

The typical simplifying assumption for large break LOCA M&E releases analyses is that main feedwater flow equals steam flow at full power operation. While performing the transient initialization checks to assure that the correct steam flow was achieved after changing the initial steam pressure in the SGs, the applicant discovered that the value that was used in the previous analysis for the main feedwater flow rate for each SG was about 50 percent too low. Correction of the initial feedwater flow rate input value increases the stored energy in the secondary side of each SG after blowdown. This increased the long term energy release rate and the subsequent calculated containment pressure response. The staff audited the revised calculation, and agreed with the applicant that the effect on the containment pressure results is small. The applicant has corrected the value and the staff finds the revised feedwater flow acceptable.

g. Equilibration Temperature

The long term LOCA M&E release methodology uses a temperature “setpoint” to define the end point for the removal of the stored energy remaining in the primary system metal, the secondary system metal and secondary side system fluid inventory. The equilibration temperature that was used in the previous analysis was higher than the value recommended in Section 14 of WCAP-15846. Reducing the equilibration temperature would increase the long-term LOCA M&E releases and increase the subsequent calculated containment pressure response. During the audits, the staff reviewed the long-term M&E release spreadsheet calculation in which this temperature input is used. The staff confirmed that the appropriate value was used. As such, the staff finds the change acceptable.

23.Y.3.2 **Inorganic Zinc Coatings within LOCA Zone of Influence and Maximum Flood Elevation inside Containment**

The applicant previously made a design change to reduce the potential post-LOCA amount of debris inside containment to support Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR Sump Performance” evaluations. This design change replaces the inorganic zinc coatings on surfaces inside containment that are within the zones of influence of a LOCA and within the flood elevation (this does not include the containment vessel) with epoxy. To reflect this change in the CEM input, the applicant replaced the inorganic zinc coating material properties for all of the internal heat sinks in the CEM with those of epoxy paint.

Epoxy has lower volumetric specific heat and thermal conductivity values than inorganic zinc (Table 6.2.1.1-8 of the DCD). The staff finds that replacing the inorganic zinc coating material properties for all of the internal heat sinks in the CEM with those of epoxy paint is conservative and acceptable.

23.Y.3.3 **Specific Heat for the Inorganic Zinc and Epoxy Coatings**

Table 6.2.1.1-8 of the DCD and Table 13-49 of WCAP-15846 list the thermal properties for the various materials that are used in the CEM. The applicant replaced the specific heat values for the inorganic zinc and epoxy coatings materials with the lowest values provided by the vendors.

The staff finds that using the lowest specific heat values for the inorganic zinc and epoxy coatings materials in the CEM is conservative and acceptable.

23.Y.3.4 **Material Properties for the Containment Shell**

Section 6.2.1.1.2 of the DCD states that the containment vessel is designed and constructed in accordance with the ASME Code, Section III, Subsection NE, “Metal Containment,” as described in Section 3.8.2. The applicant updated the containment vessel material properties (density, specific heat, thermal conductivity) to reflect the values specified by the ASME Code and biased for minimum heat removal in the CEM (Table 13-136 of WCAP-15846).

The staff finds that using the ASME Code-specified material properties are consistent with the design of the containment vessel and biasing the property values is conservative and acceptable.

23.Y.3.5 **Heat Transfer from Containment Vessel Shell below Operating Deck**

A portion of the containment shell below the operating deck is incorrectly modeled in the previous CEM as an internal heat sink with heat transfer from both inside and outside surfaces. To correct the incorrect modeling, the applicant proposed to model the surface as insulated, transferring no heat from the outside surface.

The staff finds the above model correction conservative because it ignores any heat transfer to the outside through the portion of containment shell of interest and, therefore, acceptable.

23.Y.3.6 **Crediting Selected, Existing Thermal Conductors**

The applicant has changed the CEM to credit a few of the many existing heat structures in containment that were not credited for heat removal before. This change has the effect of reducing peak containment pressure. Section 1.1 of APP-GW-GLR-096, Revision 3, states the following:

> Some of the existing thermal conductors that are in the containment evaluation model are not credited for heat and mass transfer in the peak pressure/temperature analyses because they were not part of the official design at the time the model was first developed. This includes the platforms and gratings inside containment. As shown in

Section 13 of WCAP-15846 [Reference 2], these thermal conductors were modeled as insulated on both sides to prevent heat and mass transfer to them. As described in Section 7.2, heat and mass transfer to some of these conductors have been “turned on” now in the containment evaluation model for the peak pressure/temperature analyses to help offset the impact of the higher LOCA M&E releases.

Figure 7.2-1 of APP-GW-GLR-096, Revision 3, schematically shows the locations of the existing thermal conductors that are now being credited for heat and mass transfer in the peak pressure/temperature calculations.

Heat transfer to the thermal conductors in the “dead-ended” compartments is allowed only during the initial blowdown phase. The accumulator compartments (PXS-A, PXS-B) are located below the core makeup tank compartment and are only open at the top. This inhibits the circulation of break flow to these compartments and they are described as dead-ended. Table 7.4-2 of APP-GW-GLR-096, Revision 3, lists all the heat sink thermal conductors that are currently credited in the CEM for heat transfer in each dead-ended compartment.

During the calculation audits, the staff requested further justification for crediting heat and mass transfer to the dead-ended compartments below the operating deck during the blowdown phase. In a letter dated July 7, 2011, the applicant provided a discussion of the large scale test (LST) benchmarks and the correlation to the AP1000 containment. In the LST benchmarks, dead-ended compartments were observed to be well-mixed. Although the LST volumes and flow path areas could not be directly correlated to the AP1000 volumes and flow path areas, the staff agrees with the applicant that the results are representative of the actual plant conditions. The staff also examined other experimental containment benchmarks referenced in APP-GW-GLR-096, such as the HDR and Battelle Model Containment tests to confirm that the applicant’s treatment of dead-end compartments is appropriate.

Heat transfer to certain thermal conductors in the compartments above the operating deck and in “flow-through” compartments below the operating deck is allowed for the entire transient. SG, CMT, and vertical access tunnel (VAT) compartments, which have multiple flow paths with large areas from adjacent compartments allowing flow of steam through each compartment, are describes as flow-through compartments. Figure 7.2-1 also shows the paths for circulation between the various compartments below the operating deck. The LOCA break location is assumed to be in the lower east SG compartment. Section 7.4.1 of APP-GW-GLR-096, Revision 3, provides the following reasons for assuming SG, CMT, and VAT compartments as flow-through compartments:

- The steam release is low, as the hot leg and cold legs are located at approximately the 100-ft elevation. The rising jet/plume will help circulate the containment atmosphere.

- The PCS is actuated to provide cooling by sensible heating and evaporation of water that is evenly distributed on top of the external surface of the containment vessel. This will induce falling wall plumes that will help circulate the containment atmosphere.

- The containment atmosphere is well mixed and the temperature field is not stratified during, and just after, the large LOCA blowdown occurs.

- The concrete, sump, and RCS metal heat sources are located low in the containment. The heat release from these sources will induce rising thermal plumes to help circulate the containment atmosphere.

- The SG, CMT, and VAT compartments that are located below the operating deck are connected to one another and are not dead-ended. This promotes circulation and mixing between these compartments and the volume above the operating deck.

The applicant conservatively ignored heat transfer to the floors of all the compartments because the formation of a liquid layer will inhibit condensation heat transfer to those heat sinks. The staff's SER for the CEM originally included the restriction that heat structures in dead-ended compartments below the operating deck be turned off after blowdown. There is greater uncertainty in flow paths below the operating deck and this restriction was imposed, in part, to ensure conservatism in the overall calculation. The applicant has demonstrated that there remains conservatism in the overall calculation when crediting a few of the heat structures below the operating deck. As described in Section 7.2 of APP-GW-GLR-096, Revision 3, the applicant conservatively ignored in the CEM a significant heat conductor sinks that are available in the AP1000 containment. These include about 35,000 pounds of carbon steel in the CMT and VAT compartments and close to 300,000 pounds of carbon steel above the operating deck (the jib crane, the integrated head stand, etc.). In light of the significant mass of heat structures above and below the operating deck that are not credited, the staff finds that crediting a few is acceptable. This approval does not imply additional heat structures below the operating deck will be approved by the NRC in the future.

Based on its review of APP-GW-GLR-096, Revision 3, the staff determined that crediting the existing thermal conductors for heat and mass transfer in the CEM is consistent with the guidance provided in NUREG-0800 Section 6.2.1.1.A and acceptable.

23.Y.3.7 **Release of Accumulator Nitrogen Gas after Coolant Injection**

Energy from the pressurized nitrogen gas inside the accumulators is used to force the water from the accumulators when the RCS pressure drops below 700 psig. The gas will expand into the DVI lines after all of the water from the accumulators has been discharged and collect at the high points in the RCS. Some of the gas will be released into the containment atmosphere. The previous CEM did not account for the contribution of the nitrogen gas to the containment peak pressure in the containment response analyses. The applicant proposed to update the CEM to include the effect of accumulator nitrogen gas released into the containment.

The applicant assumes an isothermal expansion of nitrogen gas from accumulators to the containment, conservatively ignoring heat transfer needed from the containment atmosphere to the nitrogen gas for such an expansion. The nitrogen gas is released directly from the accumulators to the containment atmosphere over a period of 60 seconds after the water discharge ends. Section 7.2 of APP-GW-GLR-096, Revision 3, states that "[t]he model is not sensitive to this assumption since the gas release occurs before the time of peak pressure (typically around 1500 seconds after event initiation)." The staff finds that the containment peak pressure and temperature are not sensitive to the assumed duration of nitrogen release because the peaks occur much later after the accumulators empty for double ended guillotine cold leg break LOCA and much before the accumulators empty for double ended guillotine hot leg break LOCA and MSLB accidents. The staff finds the applicant's proposed modeling of accumulator nitrogen gas released into the containment conservative and, therefore, acceptable.

As a result of making changes to the WGOTHIC AP1000 CEM inputs to address the errors and updates discussed above and the two errors stated in Section 23.X of this report, the calculated containment peak pressure increased from 398.5 kPa (57.8 psig) stated in AP1000 DCD Revision 15 to 402.0 kPa (58.3 psig) (Figure B-2 of APP-GW-GLR-096, Revision 3), which is below the containment design pressure of 406.8 kPa (59 psig). Considering the conservatisms build into the WGOTHIC AP1000 CEM and that the containment peak pressure stays below the containment design value, the staff finds that the updated containment peak pressure of 402.0 kPa (58.3 psig) is acceptable.

During the audits, the staff reviewed the applicant's calculation reports supporting the results of analyses presented in APP-GW-GLR-096, Revision 3.

Based on its review, the staff determined that after making changes to the WGOTHIC AP1000 CEM inputs, the calculated pressure is below the containment design pressure and, therefore, acceptable. The audits the staff performed confirmed these conclusions.

The staff determined that the changes to the WGOTHIC AP1000 CEM inputs did not affect the staff's conclusions as stated in NUREG-1793 and the design is compliant with GDC 50.

During its review, the staff found the applicant's proposed change to TS 5.5.8 is acceptable because it reflects the peak containment pressure following a DBA as given in DCD Tier 2 Section 6.2.

Based on its review, the staff determined that the DCD changes proposed in Appendix B of APP-GW-GLR-096, Revision 3, resulting from changes to the WGOTHIC AP1000 CEM inputs, are acceptable. The staff confirmed that the applicant has made the proposed changes in DCD, Revision 19.

23.Y.4 Conclusion

Based on its review the staff concludes that the design changes described above are acceptable and the design is compliant with GDC 50. During its review, the staff found that applicant's proposed change to TS 5.5.8 is acceptable.

24. CONCLUSION

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed Westinghouse's changes to the AP1000 design documentation (see Section 1.5 of this report). On the basis of the evaluation described in the AP1000 FSER (NUREG-1793, NUREG-1793 Supplement 1) and this report, the NRC staff concludes that the AP1000 design documentation (up to and including Revision 19 to the AP1000 design control document) is acceptable and that Westinghouse's application for design certification meets the requirements of Subpart B, “Standard Design Certifications,” of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

A. CHRONOLOGY

This appendix contains a chronological listing of routine licensing correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC) and Westinghouse regarding the review of the amendment to the AP1000 passive plant design under Project No. 711 and Docket No. 52-006. Please note that Westinghouse Technical Reports are listed with the References in Appendix B.

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
3/28/06	ML061440082	Westinghouse Slides from the EPRI/NEI I&C Workshop Held on March 28, 2006.		Westinghouse	
3/30/06	ML060690365	AP1000 Design Certification - Summaries of Telephone Calls Held Between September 29, 2005, and November 3, 2005	Quinones-Navarro, L.	NRC/NRR/ADRA/ DNRL/NRBA	
3/31/06	ML060960221	Westinghouse AP1000 COL Technical Report Submittal.	Sterdis, A.	Westinghouse	DCP/NRC1728
3/31/06	ML060960229	AP1000 Standard Combined License Technical Report, Benchmark Program for Piping Analysis Computer Programs, APP-GW-GLR-006, Revision 0.		Westinghouse	
3/31/06	ML060960239	AP1000 Standard Combined License Technical Report, Main Control Room Inleakage Testing, APP-GW-GLR-007, Revision 0.		Westinghouse	
3/31/06	ML060960252	AP1000 Standard Combined License Technical Report, Request for Closure of COL Items in DCD Chapter 11 Identification of Adsorbent Media, APP-GW-GLR-008, Revision 0.		Westinghouse	

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
3/31/06	ML060960261	AP1000 Standard Combined License Technical Report, Closure of COL Items in DCD Chapter 11 Dilution and Control of Boric Acid Discharge, APP-GW-GLR-014, Revision 0.		Westinghouse	
3/31/06	ML061030089	APP-GW-GLR-010, Rev 0, "AP1000 Standard Combined License Technical Report; AP1000 Main Control Room Staffing roles and Responsibilities."	Harmon, D. L.	Westinghouse	AP1000
3/31/06	ML061030092	APP-GW-GLR-011, Rev 0, "AP1000 Standard Combined License Technical Report; Execution and Documentation of the Human Reliability Analysis/Human Factors Engineering Integration."	Schulz, T.	Westinghouse	AP1000
3/31/06	ML061030093	APP-GW-GLR-012, Rev 0, "AP1000 Standard Combined License Technical Report; AP1000 Human Factors Engineering Program and Human System Interface Design."	Harmon, D. L.	Westinghouse	AP1000
3/31/06	ML061030102	APP-GW-GL-011, Rev 0, "AP1000 Identification of Critical Human Actions and Risk Important Tasks."	Schulz, T.	Westinghouse	AP1000
4/3/06	ML060950662	03/29/2005 Summary of Telephone Call With NuStart re Additional Information Needed for AP1000 COL Review.	Bloom, S. D.	NRC/NRR/ADRA/DNRL/NRBA	

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
4/5/06	ML060970266	AP1000 COL Design Change Review.	Sterdis, A.	Westinghouse	DCP/NRC1729
4/5/06	ML060970270	APP-GW-GLN-002, Rev. 0, "Zinc Addition."	Lindgren, D. A. Meneely, T. K. Winters, J. W.	Westinghouse	
4/5/06	ML060970272	APP-GW-GLN-003, Rev. 0, "Hydrogen Igniter Locations."	Lindgren, D. A. Mandava, R. McDermott, D. J.	Westinghouse	
4/5/06	ML061020488	NuStart Organization Update: NRC Projects No. 740 and 744.	Kray, M. C.	NuStart Energy Development, LLC	
4/6/06	ML060970267	APP-GW-GLN-001, Rev. 0, "Passive Residual Heat Removal Heat Exchanger."	Quinn, K. Wiseman, D. A.	Westinghouse	
4/6/06	ML061030082	Westinghouse, AP1000 COL Technical Report Submittal (HFE Related).	Sterdis, A.	Westinghouse	DCP/NRC1731
4/7/06	ML060690383	AP1000 Design Certification - Drop-In Visit on September 29, 2005	Quinones, L. N.	NRC/NRR/ADRA/ DNRL/NRBA	
5/16/06	ML061360164	Letter to NuStart on the Status of Westinghouse Technical Reports.	Coffin, S. M.	NRC/NRR/ADRA/ DNRL	
5/18/06	ML061430014	NuStart Bellefonte COL Project, AP-1000 Pre-Application System.	Hastings, P. S.	NuStart Energy Development, LLC	AP-1000
5/22/06	ML061440346	Submittal of APP-GW-GLR-017, Rev. 0, "AP1000 Standard Combined License Technical Report: Resolution of Common Q NRC Items."	Sterdis, A.	Westinghouse	DCP/NRC1736

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
5/22/06	ML061440381	Submittal of APP-GW-GLR-005, Rev. 0, "AP1000 Standard Combined License Technical Report: Containment Vessel Design Adjacent to Large Penetrations."	Sterdis, A.	Westinghouse	DCP/NRC1734
5/22/06	ML061440387	Submittal of APP-GW-GLN-006, Rev. 0, "Methodology for Qualifying AP1000 Safety Related Electrical & Mechanical Equipment."	Sterdis, A.	Westinghouse	DCP/NRC1735
5/22/06	ML061440479	AP1000 COL Standard Design Change Submittal. TR39	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1737
5/26/06	ML061590366	NuStart Bellefonte COL Project - AP-1000 Pre-Application Submittals.	Hastings, P. S.	NuStart Energy Development, LLC	AP1000
5/29/06	ML061520152	AP1000 COL Standard Design Change Submittal. TR46	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1741
5/29/06	ML061520154	AP1000 COL Standard Design Change Submittal. TR44	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1740
5/29/06	ML061520157	AP1000 COL Standard Design Change Submittal. TR67	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1739
5/30/06	ML061530083	AP100 COL Standard Design Change Submittal of Sensitive and Non-Sensitive Versions of APP-GW-GLR-016, Rev 0.	Sterdis, A.	Westinghouse	APP-GW-GLR-046, Rev. 0 DCP/NRC1742
5/30/06	ML061530093	APP-GW-GLR-016, Rev. 0, "AP1000 Standard Combined License Technical Report, AP1000 Pressurizer Design."		Westinghouse	DCP/NRC1742
5/30/06	ML061530485	WCAP-16361-P, Rev 0, "Westinghouse Setpoint Methodology for Protection Systems - AP1000."	Reagan, J. R. Solomon, D. K. Trozzo, R. W. Williams, T. P.	Westinghouse	AP1000

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
5/31/06	ML061530087	APP-GW-GLR-016, Rev. 0, "AP1000 Standard Combined License Technical Report, AP1000 Pressurizer Design."		Westinghouse	
5/31/06	ML061530484	AP1000 COL Technical Report Submittal.	Sterdis, A.	Westinghouse	AP1000 AW-06-2155 DCP/NRC1744
5/31/06	ML061600440	APP-GW-GLR-016, Rev. 0, "AP1000 Pressurizer Design," Technical Report Number 36.		Westinghouse	DCP/NRC1749
5/31/06	ML061600443	APP-GW-GLR-016, Rev. 0, "AP1000 Pressurizer Design," Technical Report Number 36.		Westinghouse	DCP/NRC1749
5/31/06	ML061840577	WCAP-16361-NP, Revision 0, "Westinghouse Setpoint Methodology for Protection Systems - AP1000."	Reagan, J. R. Solomon, D. K. Trozzo, R. W. Williams, T. P.	Westinghouse	
5/31/06	ML070220072	CD-ROM File: APP-GW-GLR-016, Rev. 0, "AP1000 Standard Combined License Technical Report, AP1000 Pressurizer Design."		Westinghouse	
6/5/06	ML061580451	Transmittal of AP1000 COL Standard Design Change Submittal. TR6	Sterdis, A.	Westinghouse	DCP/NRC1746
6/5/06	ML061580629	AP1000 COL Standard Design Change Submittal. TR 8	Sterdis, A.	Westinghouse	DCP/NRC1747
6/5/06	ML061600439	AP1000 COL Standard Design Change Resubmittal of Sensitive and Non-Sensitive Versions of APP-GW-016, Revision 0.	Sterdis, A.	Westinghouse	APP-GW-GLR-016, Rev. 0 DCP/NRC1749

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
6/5/06	ML062230089	AP1000 COL Standard Design Change Resubmittal of Sensitive and Non-Sensitive Versions of APP-GW-GLR-016, Revision 0.	Sterdis, A.	Westinghouse	DCP/NRC1749
6/5/06	ML062230094	AP1000 COL Standard Design Change Resubmittal of Sensitive and Non-Sensitive Versions of APP-GW-GLR-016, Revision 0.	Sterdis, A.	Westinghouse	DCP/NRC1749
6/14/06	ML061670158	AP1000 COL Standard Design Change Submittal. TR3	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1751
6/30/06	ML061840457	Transmittal of AP1000 COL Standard Technical Report Submittal. TR43	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-018, Rev 0 DCP/NRC1758
6/30/06	ML061840461	AP1000 Standard Combined License Technical Report, Failure Modes and Effects Analysis and Software Hazards Analysis for AP1000 Protection System, APP-GW-GLR-018, Rev. 0. TR43		Westinghouse	AP1000 DPC/NRC1758
6/30/06	ML061840472	Submittal of AP1000 COL Technical Report Submittal. TR45	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-027, Rev 1 DCP/NRC1753
6/30/06	ML061840473	AP1000 Standard Combined License Technical Report, Operator Actions Minimizing Spurious ADS Actuation, APP-GW-GLR-027, Revision 1.		Westinghouse	AP1000 DCP/NRC1753
6/30/06	ML061840537	AP1000 COL Standard Technical Report Submittal. TR65	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1755

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
6/30/06	ML061840539	AP1000 Standard Combined License Technical Report, Spent Fuel Storage Racks Criticality Analysis, APP-GW-GLR-029, Revision 0.		Westinghouse	AP1000 DCP/NRC1755
6/30/06	ML061840573	AP1000 COL Technical Report Submittal. TR28	Sterdis, A.	Westinghouse	DCP/NRC1752
6/30/06	ML061860405	Transmittal of Proprietary and Non-Proprietary Technical Document Information, WCAP-16592-P Rev. 0 and WCAP-16592-NP Rev. 0, "Software Hazard Analysis of AP1000 Protection and Safety Monitoring System".	Winters, J. W.	Westinghouse	AP1000 AW-06-2172 DCP/NRC1757
6/30/06	ML061860412	WCAP-16592-NP, Revision 0, "Software Hazard Analysis of AP1000 Protection and Safety Monitoring System."	Burns, J. E. Erin, L. E. Senechal, R. R. Wilson, T. C.	Westinghouse	AP1000 APP-PMS-GER-001, Rev 0 AW-06-2171 DCP/NRC1757
6/30/06	ML061860421	WCAP-16592-P, Revision 0, "Software Hazard Analysis of AP1000 Protection and Safety Monitoring System."	Burns, J. E. Erin, L. E. Senechal, R. R. Wilson, T. C.	Westinghouse	AP1000 APP-PMS-GER-001, Rev 0 AW-06-2171 DCP/NRC1757
6/30/06	ML061860491	Transmittal of Proprietary and Non-Proprietary Technical Document Information, WCAP-16438-P Rev. I and WCAP-16438-NP Rev. 1, "FMEA of AP1000 Protection and Safety Monitoring System".	Winters, J. W.	Westinghouse	AP1000 AW-06-2171 DCP/NRC1756
6/30/06	ML061860498	WCAP-16438-NP, Revision 1, "FMEA of AP1000 Protection and Safety Monitoring System."	Cook, B. A. Erin, L. E. Senechal, R. R. Wilson, T. C.	Westinghouse	AP1000 AW-06-2171

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
6/30/06	ML061860503	WCAP-16438-P, Revision 1, "FMEA of AP1000 Protection and Safety Monitoring System."	Cook, B. A. Erin, L. E. Senechal, R. R. Wilson, T. C.	Westinghouse	AP1000 APP-GW-JJ-002, Rev 1 AW-06-2171
6/30/06	ML070220073	CD-ROM File: APP-GW-GLR-021, Rev. 0, "AP1000 Standard Combined License Technical Report, AP1000 As-Build COL Information Items."		Westinghouse	
7/7/06	ML061920159	AP1000 COL Standard Approach for Reactor Design.	Sterdis, A.	Westinghouse	DCP/NRC1761
7/7/06	ML061920397	AP1000 COL Standard Technical Report Submittal. TR54	Sterdis, A.	Westinghouse	DCP/NRC1760
7/7/06	ML061920400	Enclosure 2, APP-GW-GLR-033, Revision 0 "Spent Fuel Storage Rack Structure/Seismic Analysis," Technical 54.		Westinghouse	
7/7/06	ML061920401	Enclosure 1, APP-GW-GLR-033, Revision 0, "Spent Fuel Storage Rack Structure/Seismic Analysis," Technical 54.		Westinghouse	
7/7/06	ML062230090	AP1000 COL Standard Technical Report Submittal of Sensitive and Non-Sensitive Versions of APP-GW-GLR-033, Revision 0.	Sterdis, A.	Westinghouse	DCP/NRC1760
7/7/06	ML062230098	AP1000 COL Standard Technical Report Submittal of Sensitive and Non-Sensitive Versions of APP-GW-GLR-033, Revision 0.	Sterdis, A.	Westinghouse	DCP/NRC1760

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
7/14/06	ML062010136	Westinghouse Response to NRC Regulatory Issue Summary 06-06.	Sterdis, A.	Westinghouse	DCP/NRC1762 RIS-06-006 RIS-06-009
7/17/06	ML061980343	07/27/06 Notice of Meeting with AP1000 Design-Centered Working Group (DCWG) Regarding Pre-COL Activities.	Starefos, J. L.	NRC/NRR/ADRA/DNRL	
7/17/06	ML062000089	NuStart Bellefonte COL Project - NRC Project No. 740 Response to RIS 06-06, New Reactor Standardization Needed to Support the Design-Centered Licensing Review Approach.	Kray, M. C.	NuStart Energy Development, LLC	FOIA/PA-2008-0001 RIS-06-006
7/19/06	ML062020163	AP1000 Oath of Affirmation Submittal for Previously Submitted COL Technical Reports.	Sterdis, A.	Westinghouse	DCP/NRC1759
7/19/06	ML062020216	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-022, Revision 1 (Technical Report Number 8).	Sterdis, A.	Westinghouse	DCP/NRC1763
7/20/06	ML062230088	Transmittal of AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-016 (Technical Report 36) and APP-GW-GLR-033 (Technical Report 54), Sensitive and Non-Sensitive Versions Electronically.	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-016 APP-GW-GLR-033 DCP/NRC1764
7/21/06	ML061870562	Westinghouse AP1000 - Requests for Withholding Information from Public Disclosure, AW-06-2172.	Bloom, S. D.	NRC/NRR/ADRA/DNRL	AP1000 AW-06-2172 WCAP-16592-P

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
7/21/06	ML061870615	Westinghouse, AP1000 - Request for withholding Information from Public Disclosure (AW-06-2171).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AW-06-2171 WCAP-16438-P, Rev 1
7/21/06	ML061950703	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 36 - Request for Additional Information (TAC No. MD2109).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	TAC MD2109
7/21/06	ML062000203	Westinghouse, Request for Withholding Information from Public Disclosure.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	RIS-05-006
7/24/06	ML061990552	Westinghouse, Withholding from Public Disclosure, APP-GW-GLR-016, Revision 0, "AP1000 Pressurizer Design".	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	APP-GW-GLR-016, Rev 0 RIS-05-006
7/25/06	ML061870504	AP1000, Withholding From Public Disclosure, WCAP-16361-P, "Westinghouse Setpoint Methodology for Protection Systems - AP1000" (AW-06-2155).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AW-06-2155
7/28/06	ML062120598	AP1000 COL Standard Design Change Submittal, APP-GW-S2R-010, Rev. 0, "Extension of Nuclear Island Seismic Analysis to Soil Sites," Technical Report No. 03.	Sterdis, A.	Westinghouse	AP1000 APP-GW-S2R-010, Rev 0 DCP/NRC1765

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
7/31/06	ML062130591	AP1000 COL Standard Technical Report, Submittal of Proprietary and Non-Proprietary Technical Document Information, WCAP-16620-P Rev. 0 and WCAP-16620-NP Rev. 0, "Consistency of Reactor Vessel Core Support Structure Materials Relative to Know Issues....	Sterdis, A.	Westinghouse	AW-06-2183 DCP/NRC1767 WCAP-16620-NP, Rev 0 WCAP-16620-P, Rev 0
7/31/06	ML062130594	WCAP-16620-NP, Rev 0, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant." TR12	Altman, D. A. Forsyth, D. R. Singleton, N. R. Smith, R. E. Snyder, M. D.	Westinghouse	DCP/NRC1767
7/31/06	ML062130597	WCAP-16620-P, Rev 0, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant."	Altman, D. A. Forsyth, D. R. Gilmore, C. B. Harbison, L. S. Imbrogno, G. M. Kubes, J. E. Kuenzel, A. J. Singleton, N. R. Smith, R. E. Snyder, M. D.	Westinghouse	DCP/NRC1767

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
7/31/06	ML062130602	AP1000 COL Standard Technical Report Submittal. Submittal of Proprietary and Non-Proprietary Technical Document Information, WCAP-16624-P Rev. I and WCAP-16624-NP Rev. 1, "Reactor Internals Materials Changes" TR24	Sterdis, A.	Westinghouse	DCP/NRC1766 WCAP-16624-NP, Rev 1 WCAP-16624-P, Rev 1
7/31/06	ML062130604	WCAP-16624-NP, Rev 1, "Reactor Internals Materials Changes for the AP1000 Plant."	Forsyth, D. R. Singleton, N. R.	Westinghouse	
7/31/06	ML062130609	WCAP-16624-P, Rev 1, "Reactor Internals Materials Changes for the AP1000 Plant."	Forsyth, D. R. Singleton, N. R.	Westinghouse	
8/1/06	ML062130077	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 32 - Request for Additional Information (TAC NO. MD1432)	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	TAC MD1432
8/7/06	ML062150407	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 37 - Request for Additional Information (TAC NO. MD1433).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL/NAPB	AP1000 TAC MD1433
8/7/06	ML062210422	AP1000 COL Standard Technical Report Submittal.	Sterdis, A.	Westinghouse	DCP/NRC1768
8/8/06	ML062150389	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 6 - Request for Additional Information (TAC NO. MD2174).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL/NAPB	AP1000 TAC MD2174

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8/14/06	ML063040677	NuStart Bellefonte COL Project, AP-1000 Pre-Application Submittals.	Hastings, P. S.	NuStart Energy Development, LLC	
8/21/06	ML062270662	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 36 - Request for Additional Information (TAC MD2109).	Bloom, S. D.	NRC/NRR/ADRA/ NRL	TAC MD2109
8/29/06	ML062350450	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 6 - Request for Additional Information (TAC NO. MD2174).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	TAC MD2174
8/29/06	ML062350476	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 8 - Request for Additional Information (TAC MD2175).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	TAC MD2175
8/29/06	ML062550528	08/29/06 Meeting Slides on Overview of AP1000 COL Piping Analysis, Hazard Evaluation, and LBB Evaluation.		Westinghouse	
8/29/06	ML062550529	08/29/06 Meeting Slides on LBB Regulatory Issues.	Lindgren, D. A.	Westinghouse	
8/29/06	ML062550533	08/29/06 Meeting Slides on LBB and Piping Analysis.	Kotwicki, P. J.	Westinghouse	
8/31/06	ML062430452	09/12/06 - Forthcoming Meeting Closed Meeting with Westinghouse Regarding AP1000 Reactor Fuels.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	
9/1/06	ML062340261	AP1000 - Requests for Withholding Information From Public Disclosure (AW-06-2182).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AW-06-2182

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9/7/06	ML062340224	AP1000 - Request for Withholding Information From Public Disclosure (AW-06-2183).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AW-06-2183
9/7/06	ML062490459	09/19/06 Notice of Forthcoming Closed Meeting with Westinghouse Regarding Security Issues.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	
9/8/06	ML062560035	AP1000 COL Response to Request for Additional Information (TR #32).	Sterdis, A.	Westinghouse	DCP/NRC1772 TAC MD1432
9/8/06	ML062560036	Submittal of AP1000 COL Standard Technical Report. TR62	Sterdis, A.	Westinghouse	DCP/NRC1773
9/8/06	ML062560080	AP1000 COL Standard Technical Report Submittal. TR16	Sterdis, A.	Westinghouse	DCP/NRC1774
9/11/06	ML062500184	Westinghouse AP1000 Combined License (COL) Pre-application Technical Report 8 - Request for Additional Information (TAC NO. MD2175).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL/NAPB	TAC MD2175
9/12/06	ML062620374	09/12/06 Meeting Slides, "AP1000 COL Fuel Technical Report," Pre-submittal Meeting with NRC.		BNFL, Inc. Westinghouse	
9/12/06	ML062620382	09/12/06 Meeting Slides, "AP1000 COL Fuel Technical Report," Pre-submittal Meeting with NRC.		BNFL, Inc. Westinghouse	
9/12/06	ML062850171	Enclosure 4 - Transmittal of Non-Proprietary Presentation Material for AP1000 COL Fuel Technical Report Pre-Submittal Meeting with NRC.	Lindgren, D. A. Winters, J. W.	Westinghouse	AP1000 AW-06-2198

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9/12/06	ML062850174	Enclosure 3 - Transmittal of Proprietary Presentation Material for AP1000 COL Fuel Technical Report Pre-Submittal Meeting with NRC.	Lindgren, D. A. Winters, J. W.	Westinghouse	AP1000 AW-06-2198
9/18/06	ML062560269	08/10/06 Summary of Category 1 Meeting with Westinghouse to Discuss the AP1000 Technical Reports Related to Seismic Analyses and Structures.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	
9/18/06	ML062640426	09/20/06 AP1000 DCWG Meeting Handout - Draft ESBWR FSAR Examples.		NRC/NRR	
9/19/06	ML062500207	10/03-05/06 Notice of Meeting with Westinghouse Regarding Instrumentation and Control for the AP1000.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	
9/19/06	ML062840523	09/19/06 Meeting Slides, Westinghouse AP1000 Security Overview.		Westinghouse	
9/19/06	ML062840530	09/19/06 Meeting Slides, Westinghouse NSIR AP1000 Overview.		Westinghouse	
9/20/06	ML062640422	09/20/06 AP1000 DCWG MTG Handout - Slides.		NuStart Energy Development, LLC	
9/20/06	ML062640427	09/20/06 AP1000 DCWG MTG Handout - Categories of Side Heads Drafts.		NRC/NRR	
9/21/06	ML062680032	AP1000 COL Standard Technical Report Submittal.TR71B	Sterdis, A.	Westinghouse	DCP/NRC1778
9/22/06	ML062680030	AP1000 COL Standard Technical Report Submittal. TR71A	Sterdis, A.	Westinghouse	DCP/NRC1781

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9/22/06	ML062690205	AP1000 COL Standard Technical Report Submittal. TR23	Sterdis, A.	Westinghouse	DCP/NRC1782
9/28/06	ML062700575	Westinghouse AP1000 Combined License (COL) Pre-application Technical Report 32 - Request for Additional Information (TAC No. MD1432).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	TAC MD1432
9/29/06	ML062700538	AP1000 - Requests for Withholding Information from Public Disclosure (AW-06-2198).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AW-06-2198
9/29/06	ML062760234	AP1000 COL Standard Technical Report Submittal. TR33	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLN-009, Rev. 0 DCP/NRC1785
10/2/06	ML062690162	Safety Evaluation Regarding AP1000 COL Technical Report 37 Related to Hydrogen Igniter Locations (TAC No.: MD1433).	Dennig, R. L.	NRC/NRR/ADES/DSS/ SCVB	TAC MD1433
10/4/06	ML062850159	Transmittal of Proprietary and Non-Proprietary Presentation Material for AP1000 COL Fuel Technical Report Pre-Submittal Meeting with NRC, September 12, 06.	Winters, J. W.	Westinghouse	AP1000 AW-06-2198 DCP/NRC1788
10/5/06	ML062690217	08/29-30/06 Summary of Category 1 Meeting with Westinghouse Regarding AP1000 Technical Reports Related to Leak Before Break and Piping Analysis.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL/NAPB	AP1000
10/5/06	ML062690228	09/12/06 Summary of Closed Meeting with Westinghouse Regarding the AP1000 Technical Reports Related to Fuel.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL/NAPB	

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10/5/06	ML062850037	G060834 - D. Klein Memo Re: Short Summary for Meeting with Westinghouse on October 17, 06.	Klein, D. E.	NRC/Chairman	G060834
10/6/06	ML062790392	Request for Additional Information Regarding Westinghouse AP1000 Pre-Combined Operating License Application Work Technical Report APP-GW-GLR-027, Revision 1, "Operator Actions Minimizing Spurious Automatic Depressurization System (ADS) Actuation"	Weerakkody, S. D.	NRC/NRR/ADRA/DRA/ AFPB	TAC MD2495
10/6/06	ML062850099	AP1000 COL Standard Technical Report Submittal.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1790
10/6/06	ML062850100	AP1000 COL Standard Technical Report Submittal.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1789
10/12/06	ML062890149	Transmittal of Proprietary and Non-Proprietary Presentation Material for AP1000 PMS Planning Activities Phase 1 NRC Inspection Meeting with NRC, October 3, 4, and 5, 06.	Sterdis, A.	Westinghouse	AP1000 AW-06-2208 DCP/NRC1787
10/12/06	ML062890150	Enclosure 4 - Transmittal of Non-Proprietary Presentation Material for AP1000 PMS Planning Activities Phase 1 NRC Inspection APP-PMS-GLY-001-NP.		Westinghouse	AP1000 APP-PMS-GLY-001-NP AW-06-2208 DCP/NRC1787
10/12/06	ML062890153	Enclosure 3 - Transmittal of Proprietary Presentation Material for AP1000 PMS Planning Activities Phase 1 NRC Inspection APP-PMS-GLY-001.		Westinghouse	AP1000 APP-PMS-GLY-001 AW-06-2208 DCP/NRC1787

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10/13/06	ML062860042	10/24/06-Forthcoming Joint Meeting with AP1000 and ESBWR DCWGs Regarding Pre-COL Activities.	Kevern, T. A.	NRC/NRR/ADRA/ DNRL/NESB	
10/16/06	ML062840413	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 36 - Request for Additional Information (TAC NO. MD2109).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL/NAPB	TAC MD2109
10/16/06	ML062850524	11/07-09/06 - Forthcoming Meeting with Westinghouse Regarding Human Factors Engineering For the AP1000.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AP1000
10/19/06	ML062980276	AP1000 COL Standard Technical Report Submittal. TR85	Sterdis, A.	Westinghouse	DCP/NRC1791
10/19/06	ML063250311	AP1000RCP-06-009-NP, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel," (Non-Proprietary).		Curtiss-Wright Electro-Mechanical Corp	DCP/NRC1802
10/19/06	ML063250314	AP1000RCP-06-009-P, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel," (Proprietary).		Curtiss-Wright Electro-Mechanical Corp	DCP/NRC1802
10/20/06	ML062970113	AP1000 Piping Design Acceptance Criteria Completion and Deletion.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1793
10/24/06	ML062980485	Handouts, AP1000 DCWG, Public Category 2 Meeting, NuStart Energy, TSC and Leak Rate Testing.		Duke Energy Corp NuStart Energy Development, LLC Progress Energy Co South Carolina Electric and Gas Co Westinghouse	

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10/24/06	ML062980504	List of Attendees, AP1000 DCWG Public Category 2 Meeting, 24 October 06.		NRC/NRR/ADRA/ DNRL	
10/30/06	ML062910210	AP1000 - Requests For Withholding Information From Public Disclosure (AW-06-2208).	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	AW-06-2208
10/31/06	ML062910491	10/03-05/06 Summary of Open and Closed Meeting with Westinghouse to Discuss AP1000 Instrumentation and Controls.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
10/31/06	ML063060045	Transmittal of WCAP-16652-NP (Technical Report 18) Rev. 0, "AP1000 Core & Fuel Design".	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1794 WCAP-16652-NP, Rev. 0
10/31/06	ML063060046	WCAP-16652-NP, Rev 0, "AP1000 Core & Fuel Design Technical Report."	Knott, R. P. Miller, R. W. Ray, S. Riggs, M. J. Sisk, R. B.	Westinghouse	APP-GW-GLR-059
11/1/06	ML063070473	Transmittal of AP1000 COL Standard Technical Report, APP-GW-GLR-040, Rev. 0, "AP1000 Standard Combined License Technical Report, Plant Operations, Surveillance, and Maintenance Procedures." TR70	Sterdis, A.	Westinghouse	DCP/NRC1783
11/13/06	ML063210350	Westinghouse Transmittal of Proprietary and Non-Proprietary Presentation Material for AP1000 HFE DCD Chapter 18 Meeting with the NRC on November 7, 8 and 9, 06.	Sterdis, A.	Westinghouse	DCP/NRC1795

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11/13/06	ML063210351	Westinghouse, Enclosure 4, NRC Human Factors Engineering Program Presentation (Non-Proprietary) APP-OCS-GLY-001-NP on November 7th through November 9th, 06.		NuStart Energy Development, LLC Westinghouse	AW-06-2212
11/13/06	ML063210353	Westinghouse, Enclosure 3, NRC Human Factors Engineering Program Presentation (Proprietary) APP-OCS-GLY-001-P on November 7th Through November 9th, 06.		NuStart Energy Development, LLC Westinghouse	AW-06-2212
11/14/06	ML063180085	Summary of Meeting with AP1000 and ESBWR DCWGs Regarding Pre-COL Activities.	Kevern, T. A.	NRC/NRO/DNRL	
11/14/06	ML063210447	AP1000 COL Standard Technical Report Submittal of Sensitive and Non-Sensitive Versions of APP-GW-GLN-014, Revision 0. TR61	Sterdis, A.	Westinghouse	DCP/NRC1798
11/15/06	ML063210420	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-013, Revision 0. TR30	Sterdis, A.	Westinghouse	APP-GW-GLN-013, Rev 0 DCP/NRC1800
11/16/06	ML063250056	Westinghouse AP1000 Rev. 15 DCD Table 1.8-2 Database Excerpt	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1801
11/17/06	ML063250298	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-016, Revision 0. TR34	Sterdis, A.	Westinghouse	DCP/NRC1802

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11/21/06	ML063240260	12/12-15/06 Notice of Forthcoming Meeting with Westinghouse Regarding Seismic Analyses for the AP1000.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
11/22/06	ML063070333	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 59 - Request for Additional Information (TAC No. MD1435).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD1435 WCAP-16555
11/22/06	ML063240566	Westinghouse, Withholding From Public Disclosure, APP-GW-GLN-014, Revision 0, "AP1000 Integrated Head Package".	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	APP-GW-GLN-014, Rev 0
11/22/06	ML063310048	Correction to AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-016, Revision 0. TR34	Sterdis, A.	Westinghouse	APP-GW-GLN-016, Rev 0 DCP/NRC1804
11/24/06	ML063240208	12/07/06 Forthcoming Meeting with AP1000 and ESBWR Design-Centered Working Groups Regarding Pre-COL Activities.	Kevern, T. A.	NRC/NRO/DNRL	
11/27/06	ML062910275	11/27/06 Summary of Closed Meeting With Westinghouse To Discuss AP1000 Security Issues.	Bloom, S. D.	NRC/NRR/ADRA/ DNRL	
11/29/06	ML063350051	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-010, Revision 0. TR35	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1803
11/30/06	ML063210450	APP-GW-GLN-014-NS, Revision 0, "AP1000 Integrated Head Package" Technical Report 61.		Westinghouse	DCP/NRC1798

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11/30/06	ML063210462	APP-GW-GLN-014, Revision 0, "AP1000 Integrated Head Package" Technical Report 61.		Westinghouse	DCP/NRC1798
12/4/06	ML063340562	AP1000 Requests for Withholding Information from Public Disclosure (AW-06-112), AP1000RCP-06-009, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel".	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	AP1000 AP1000RCP-06-009 AW-06-112
12/4/06	ML063380489	Safety Evaluation Regarding Westinghouse AP1000 Standard Combined License Technical Report for the New Fuel Storage Rack Criticality Analysis.	Cranston, G. V.	NRC/NRR/ADES/DSS/ SBWB	TAC MD2103
12/5/06	ML063330537	Westinghouse AP1000 Combined License Pre-Application Technical Report 3 - Request for Additional Information.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	APP-GW-S2R-010, Rev 0 TAC MD2358
12/5/06	ML063340669	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 6 - Request for Additional Information (TAC NO. MD2174).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2174
12/5/06	ML063390648	AP-1000 Spent Fuel Storage Racks Criticality Analysis.	Landry, R. R.	NRC/NRR/ADES/DSS/ SNPB	TAC MD2492
12/6/06	ML063400351	12/18/06 Notice of Forthcoming Meeting with Westinghouse Regarding Piping Design Acceptance Criteria for the AP1000.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	AP1000

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12/7/06	ML063170166	10/24/06 Summary of Category 2 Meeting with AP1000 Design-Centered Working Group, Regarding Pre-Col Activities.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	RG-1.163
12/7/06	ML063450058	12/07/06 - Meeting Handout 1 - AP1000-ESBWR Design Centered Work Group Slides.		Dominion NuStart Energy Development, LLC	AP1000
12/12/06	ML063480072	AP1000 COL Response to Request for Additional Information (TR #32).	Sterdis, A.	Westinghouse	APP-GW-GLN-002, Rev 0 DCP/NRC1809 RAI-TR32-007 TAC MD1432
12/13/06	ML063520160	Submittal of AP1000 COL Standard Technical Report, and Proprietary and Non-Proprietary Technical Document Information, WCAP-16674-P Rev. 0 and WCAP-16674-NP Rev. 0, "AP1000 I&C Data Communication and Manual Control of Safety Systems and Components." TR88	Sterdis, A.	Westinghouse	DCP/NRC1808 WCAP-16674-NP Rev 00 WCAP-16674-P Rev 00
12/14/06	ML063250158	AP1000 - Requests for Withholding Information from Public Disclosure (AW-06-2212), Presentation material for AP1000 HFE DCD Chapter 18 Meetings with the NRC, November 7, 8, and ryc, 06.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	AW-06-2212
12/15/06	ML063530015	AP1000 COL Standard Technical Report Number 49 Submittal.	Sterdis, A.	Westinghouse	DCP/NRC1810
12/15/2006	ML063530216	Subject: AP1000 COL Standard Technical Report Number 74A Submittal.	Sterdis, A.	Westinghouse	DCP/NRC1807

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12/18/2006	ML070040271	12/18/2006 - Viewgraphs from Meeting with NRC Regarding AP1000 Piping Design Acceptance Criteria.	Sterdis, A. L.	Westinghouse	
12/21/2006	ML070040379	New Reactor Licensing Activities as of December 21, 2006.		NRC/NRO/DNRL	
1/9/2007	ML063540143	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 39 - Request for Additional Information (TAC MD1849).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD1849
1/9/2007	ML063610203	11/07-09/2006 Summary of Open and Closed Meeting with Westinghouse to Discuss AP1000 Human Factors Engineering.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
1/9/2007	ML063620055	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 6 - Request for Additional Information (TAC MD2174).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2174
1/9/2007	ML070090075	01/26/2007 Notice of Forthcoming Meeting with Westinghouse and General Electric Company Regarding the Qualification Process for Electronic Submission of the Design Control Document.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
1/9/2007	ML070180114	CD-ROM file: COL Application formatting.		- No Known Affiliation	
1/12/2007	ML070170176	AP1000 COL Standard Technical Report Submittal.	Sterdis, A.	Westinghouse	DCP/NRC1812
1/16/2007	ML063180156	Westinghouse AP1000 Technical Reports.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	

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1/17/2007	ML063560013	AP1000 Requests for Withholding Information from Public Disclosure (AW-06-2222).	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	AW-06-2222 WCAP-16674-P, Rev. 0
1/17/2007	ML070090131	12/18/2006 Summary of Open Meeting with Westinghouse to Discuss AP1000 Piping Design Acceptance Criteria (DAC).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
1/18/2007	ML070170134	Safety Evaluation of Zinc Injection to AP1000 (TAC NO. MD1432).	Landry R R	NRC/NRR/ADES/DSS/ SNPB	TAC MD1432
1/18/2007	ML070240432	AP1000 COL Response to Request for Additional Information (TR #3).	Sterdis, A.	Westinghouse	DCP/NRC1814
1/24/2007	ML070250111	Request for Additional Information for Westinghouse Technical Reports (TR) 71A-B (TAC NOS.: MD3056 and MD3057).	Hamzehee, H. G.	NRC/NRR/ADES/DE/ EQVB	TAC MD3056 TAC MD3057
1/24/2007	ML090771212	E-Mail from Nilesh Chokshi, NRO to Robert B. Whorton, Subject Re: AP1000/NRC Meeting- January 31.	Chokshi, N. C.	NRC/NRO/DSER	FOIA/PA-2009-0035
1/26/2007	ML070300120	AP1000 COL Standard Technical Report Submittal, Revision 0.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC182O
1/29/2007	ML070120250	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 36 - Request for Additional Information (TAC NO. MD2109).	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	DCP/NRC 1749 DCP/NRC 1769 TAC MD2109
1/29/2007	ML070330131	AP1000 COL Response to Request for Additional Information (TR #59).	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1819 TAC MD1435

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1/29/2007	ML070330590	AP1000 COL Response to Request for Additional Information (TR #3).	Sterdis, A.	Westinghouse	AP1000 APP-GW-S2R-010, Rev 0 DCP/NRC1822
1/30/2007	ML070330064	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-021, Revision 0.	Sterdis, A.	Westinghouse	DCP/NRC1821
1/31/2007	ML070310188	12/07/2006 Summary of Meeting to Discuss Pre-Combined License Issues.	Starefos, J. L.	NRC/NRO/DNRL/ AP1000B1	
2/1/2007	ML070330588	Westinghouse, AP1000 COL Standard Technical Report Submittal.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1825
2/2/2007	ML070320299	02/21-22/2007 Notice of Forthcoming Meeting with Westinghouse Regarding Instrument and Controls for the AP1000.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
2/5/2007	ML070090190	12/12-14/2006 Summary of the Open and Closed Meeting with Westinghouse to Discuss AP1000 Seismic Analyses.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
2/8/2007	ML070430288	AP1000 COL Standard Technical Report Submittal. TR24	Sterdis, A.	Westinghouse	DCP/NRC1824
2/8/2007	ML070430364	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-018, Revision 0. TR43	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1826
2/12/2007	ML070450141	Submittal of AP1000 COL Standard Technical Report APP-GW-GLR-013, Rev 0, "Safety Class Piping Design Specifications and Design Reports Summary."	Sterdis, A.	Westinghouse	DCP/NRC1829

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2/13/2007	ML070460603	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-012, Revision 0. TR29	Sterdis, A.	Westinghouse	APP-GW-GLN-012, Rev 0 DCP/NRC1830 WCAP-16716-NP, Rev 0
2/13/2007	ML070460604	Enclosure 1, APP-GW-GLN-012, Revision 0, "AP1000 Reactor Internals Design Changes" Technical Report 29, WCAP-16716-NP.		Westinghouse	DCP/NRC1830
2/13/2007	ML070520285	Standard Technical Report Submittal AP-TR-NS01, "Containment Leak Rate Test Program."	Kray, M. C.	NuStart Energy Development, LLC	
2/13/2007	ML070520285	Standard Technical Report Submittal AP-TR-NS01, "Containment Leak Rate Test Program."	Kray, M. C.	NuStart Energy Development, LLC	
2/14/2007	ML070470034	AP1000 COL Standard Technical Report Submittal of WCAP-16696-P, Rev. 0 and WCAP-16696-NP, Rev. 0, "Strategy for Closure of the AP1000 Design Control Document, Chapter 18 Human Factors Engineering Combined Operating License Information Items." TR90	Sterdis, A.	Westinghouse	DCP/NRC1831
2/14/2007	ML070510529	Submittal of Proprietary and Non-Proprietary Technical Document Information, WCAP-16675-P, Rev. 0 and WCAP-16675-NP, Rev. 0, "Protection System Architecture". TR89	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1817 WCAP-16675-NP WCAP-16675-P
2/15/2007	ML070460228	Westinghouse AP1000 Design Change Technical Report No. 61.	Dixon-Herrity, J. L.	NRC/NRO/DE	TAC MD3607

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2/15/2007	ML070640341	Updated Administrative List of AP1000 and ESBWR DCWG Action Items.	Borsh R	Dominion	
2/16/2007	ML070520174	AP1000 COL Standard Technical Report Submittal.	Sterdis, A.	Westinghouse	DCP/NRC1833
2/16/2007	ML070520281	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-007, Revision 1. TR27	Sterdis, A.	Westinghouse	DCP/NRC1832
2/16/2007	ML070520284	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-003, Revision 1. TR37	Sterdis, A.	Westinghouse	DCP/NRC1818
2/19/2007	ML070530202	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-074, Revision 0. TR74B	Sterdis, A.	Westinghouse	DCP/NRC1835
2/19/2007	ML070580467	Transmittal of Proprietary and Non-Proprietary Presentation Material for Meeting with NRC re Instrumentation and Controls (I&C) for the AP1000, February 21 and 22, 2007.	Sterdis, A.	Westinghouse	AW-07-2245 DCP/NRC1834
2/21/2007	ML070580469	APP-PMS-GLY-002, "AP1000 I&C NRC Meeting, February 21 & 22, 2007."		Westinghouse	
2/21/2007	ML070580470	APP-PMS-GLY-002, "AP1000 I&C NRC Meeting, February 21 & 22, 2007."		Westinghouse	
2/23/2007	ML070570441	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-073, Revision 0. TR93	Sterdis, A.	Westinghouse	DCP/NRC1838

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2/26/2007	ML070330582	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 12 - Request for Additional Information (TAC NO. MD2694).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2694 WCAP-16620-P, Rev 0
2/26/2007	ML070400030	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Reports 71A and 71B - Request for Additional Information (TAC NOS. MD3056 and MD3057).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	APP-GW-GLR-037, Rev 0 APP-GW-GLR-038, Rev 0 TAC MD3056 TAC MD3057
2/28/2007	ML070470038	WCAP-16696-NP, Rev. 0, "Strategy for the Closure of the AP1000 Design Control Document, Chapter 18 Human Factors Engineering Combined Operating License Information Items."	Hunton, P. J.	Westinghouse	APP-GW-GLR-090
2/28/2007	ML070470042	WCAP-16696-P, Rev. 0, "Strategy for the Closure of the AP1000 Design Control Document, Chapter 18 Human Factors Engineering Combined Operating License Information Items."	Hunton, P. J.	Westinghouse	APP-GW-GLR-090
2/28/2007	ML070510532	WCAP-16675-NP, Rev 0, "AP1000 Protection and Safety Monitoring System Architecture Technical Report."	Drake, A. P. Hayes, T. P. Vitalbo, C. A.	Westinghouse	AP1000 APP-GW-GLR-071 DPC/NRC1817
2/28/2007	ML070510536	WCAP-16675-P, Rev 0, "AP1000 Protection and Safety Monitoring System Architecture Technical Report."	Drake, A. P. Hayes, T. P. Vitalbo, C. A.	Westinghouse	AP1000 APP-GW-GLR-071 DCP/NRC1817

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2/28/2007	ML071380082	WCAP-16650-NP, Revision 0, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines."	Corletti, M. M. Schwab, E. K.	Westinghouse	DCP/NRC1889
2/28/2007	ML071380083	WCAP-16651-NP, Revision 0, "Probabilistic Evaluation of Turbine Valve Test Frequency."	Corletti, M. M. Haessler, R. L.	Westinghouse	DCP/NRC1889
2/28/2007	ML071380085	WCAP-16650-P, Rev. 0, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines."	Corletti, M. M. Schwab, E. K.	Westinghouse	DCP/NRC1889
2/28/2007	ML071380086	WCAP-16651-P, Rev 0, "Probabilistic Evaluation of Turbine Valve Test Frequency."	Corletti, M. M. Haessler, R. L.	Westinghouse	DCP/NRC1889
3/1/2007	ML070670317	AP1000 Design-Centered Working Group, Seismic Discussions.		NuStart Energy Development, LLC	AP1000
3/1/2007	ML070670321	Bellefonte, Units 3 and 4, Westinghouse AP1000.		NuStart Energy Development, LLC	
3/1/2007	ML070670322	William States Lee Nuclear Station, Westinghouse AP1000 Units 1 and 2.		Duke Energy Carolinas, LLC	AP1000
3/1/2007	ML070670323	Virgil C. Summer, Units 2 and 3, (Westinghouse AP1000).		South Carolina Electric and Gas Co	AP1000
3/1/2007	ML070670330	AP1000 Structural Evaluation of Hard Rock Sites.	Tunon-Sanjur, L.	Westinghouse	AP1000 AP600
3/1/2007	ML070670333	Vogtle AP1000 Seismic Assessment Status.	Moore, D.	Southern Nuclear Operating Co, Inc.	AP1000
3/2/2007	ML070650347	Westinghouse, AP1000 Design-Centered Working Group Meeting on February 15, 2007.	Sterdis, A.	Westinghouse	DCP/NRC1839

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3/6/2007	ML070670260	AP1000 and ESBWR DOWA Pre-Application Interaction Topics.	Hastings, P. S.	Duke Energy Carolinas, LLC	
3/8/2007	ML070790610	NuStart Bellefonte COL Project - AP-1000 Pre-Application Submittals.	Hastings, P. S.	NuStart Energy Development, LLC	
3/9/2007	ML070720521	AP1000 COL Standard Technical Report Submittal. TR112	Sterdis, A.	Westinghouse	DCP/NRC1841
3/9/2007	ML070720536	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-020. Revision 0. TR100	Sterdis, A.	Westinghouse	DCP/NRC1842
3/9/2007	ML070920506	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-068, Revision 0. TR95	Sterdis, A.	Westinghouse	DCP/NRC 1850
3/9/2007	ML070920508	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-067, Revision 0. TR96	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-067, Rev 0 DPC/NRC1851
3/9/2007	ML070930078	AP 1000 COL Standard Technical Report Submittal of APP-GW-GLR-062, Revision 1. TR49	Sterdis, A.	Westinghouse	DCP/NRC1852
3/10/2007	ML070510374	AP1000 Requests for Withholding Information from Public Disclosure (AW-07-2242).	Bloom, S. D.	NRC/NRO/DNRL	AP1000 AW-07-2242
3/10/2007	ML070510638	AP1000 Requests for Withholding Information from Public Disclosure (AW-07-2241).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	AW-07-2241

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3/10/2007	ML070570591	Westinghouse, Withholding From Public Disclosure, "Transmittal of Proprietary Presentation Material Regarding Instrumentation and Controls (I&C) for the AP1000 Meeting with NRC, February 21 and 22, 2007."	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	AW-07-2245
3/11/2007	ML070540106	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 31 - Request for Additional Information (TAC NO. MD2695).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2695
3/11/2007	ML070570563	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 43 - Request for Additional Information (TAC MD2496).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2496
3/12/2007	ML070730653	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-075, Revision 0.	Sterdis, A.	Westinghouse	APP-GW-GLN-075 DCP/NRC 1844
3/14/2007	ML070710574	Safety Evaluation Input Regarding AP1000 COL Technical Report 36 Related to AP1000 Pressurizer Design (TAC No. MD2109).	Dennig, R. L.	NRC/NRR/ADES/DSS	TAC MD2109
3/14/2007	ML070750100	Correction to AP1000 COL Standard Technical Report Submittals of Rev. 1 to APP-GW-GLR-062, Rev. 0 to APP-GW-GLR-067 and Rev. 0 to APP-GW-GLR-068.			

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3/15/2007	ML070780076	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-054, Revision 0. TR11g			
3/15/2007	ML070780101	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-020, Revision 0. TR100			
3/15/2007	ML070780427	Westinghouse AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-022, Revision 2. TR8			
3/20/2007	ML070800483	AP1000 COL Standard Combined License Technical Report Submittal of APP-GW-GLR-091, Rev. 0. TR91			
3/25/2007	ML070750025	02/21/2007 - 02/22/2007 Summary of Meeting with Westinghouse to Discuss AP10000 Instrumentation and Control.			
3/26/2007	ML070800740	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 42 and 88 - Request for Additional Information (TAC Nos. MD1850 and MD3831).			
3/28/2007	ML070890298	Westinghouse Requesting Approval for the Use of Encryption Software for Electronic Transmission of Safeguards Information.			

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3/29/2007	ML070790327	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Reports 54 - Request for Additional Information (TAC NOS. MD2551).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2551
3/29/2007	ML070850160	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 61 - Request for Additional Information (TAC No. MD3607).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	APP-GW-GLR-014, Rev 0 TAC MD3607
3/29/2007	ML070870672	04/11-12/2007 Notice of Meeting with Westinghouse Regarding Procedures for the AP1000.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
3/29/2007	ML070940092	AP1000 COL Standard Technical Report Submittal of Proprietary and Non-Proprietary WCAP-16687-P, Rev. 1 and WCAP-16687-NP, Rev. 1.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1856
3/30/2007	ML070670301	02/15/2007 Summary of Category 1 Meeting with Westinghouse Regarding Probabilistic Risk Assessment Status for the AP1000 Reactor Design.	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	
3/30/2007	ML070800573	Westinghouse AP1000 Combined License (COL) Pre-application Technical Report 45 - Request for Additional Information (TAC NO. MD2495)	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2495

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3/31/2007	ML070940094	WCAP-16687-NP, Revision 1, "AP1000 Reactor Internals Expected and Acceptable Responses During Preoperational Vibration Measurement Program." TR10	Altman, D. A. Bhandari, D. R. Forsyth, D. R. Good, B. F. Imbrogno, G. M. Kubes, J. E. Law, Y. K. Schwirian, R. E. Singleton, N. R. Snyder, M. D. Yang, J. Yu, C.	Westinghouse	AP1000 DCP/NRC1856
3/31/2007	ML070940097	WCAP-16687-P, Revision 1, "AP1000 Reactor Internals Expected and Acceptable Responses During Preoperational Vibration Measurement Program." TR10	Altman, D. A. Bhandari, D. R. Forsyth, D. R. Good, B. F. Imbrogno, G. M. Kubes, J. E. Law, Y. K. Schwirian, R. E. Singleton, N. R. Snyder, M. D. Yang, J. Yu, C.	Westinghouse	AP1000 DCP/NRC1856
4/2/2007	ML070920133	4/16-20/2007 Notice of Forthcoming Meetings with Westinghouse Regarding Seismic Analyses for the AP1000.	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	

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4/4/2007	ML070990593	Submittal of Standard Technical Reports, AP-TR-NS01, Rev. 1, "Containment Leak Rate Test Program Description," AP-TR-NS02, Rev. 1, "Reactor Vessel Material Surveillance Program Description," and AP-TR-NS03, Rev. 1, "Equipment Qualification Program....	Kray, M. C.	NuStart Energy Development, LLC	
4/5/2007	ML070990251	Westinghouse AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-045, Revision 0. TR57	Sterdis, A.	Westinghouse	DCP/NRC1847
4/6/2007	ML070920438	Westinghouse, RAI, AP1000 Combined License (COL) Pre-Application Technical Report 44 (TAC MD2104).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	TAC MD2104
4/9/2007	ML071000291	Results of the Health Physics Branch's Review of Westinghouse AP1000 Standard Combined License Technical Report TR-36 (AP1000 Pressurizer Design).	Frye, T. J.	NRC/NRR/ADRO/ DIRS/IHPB	TAC MD2109
4/9/2007	ML071010114	AP1000 COL Response to Request for Additional Information (TR #54).	Sterdis, A.	Westinghouse	DCP/NRC1860
4/10/2007	ML071010532	AP1000 COL Response to Request for Additional Information (TR #54).	Sterdis, A.	Westinghouse	APP-GW-GLR-033, Rev 0 DCP/NRC1861 TAC MD2551
4/10/2007	ML071010536	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-098, Revision 0.	Sterdis, A.	Westinghouse	DCP/NRC1863

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4/11/2007	ML071100210	04/11/2007, Meeting Slides - AP1000 Plant Operating Procedures.	Lindgren, D. A. Long, S. Williams, M.	Westinghouse	
4/13/2007	ML071060327	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-053, Revision 0.	Hutchings, D. F. Sterdis, A.	Westinghouse	DCP/NRC1864
4/13/2007	ML071060328	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-102, Rev. 0.TR102	Hutchings, D. F. Sterdis, A.	Westinghouse	APP-GW-GLR-102, Rev. 0 DCP/NRC1862
4/13/2007	ML071060332	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-064, Revision 1.TR74a	Hutchings, D. F. Sterdis, A.	Westinghouse	DCP/NRC1865
4/19/2007	ML071060336	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 28 - Request for Additional Information (TAC MD2126).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	TAC MD2126
4/19/2007	ML071070236	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 35 - Request for Additional Information (TAC NO. MD3727).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	AP1000 TAC MD3727
4/19/2007	ML071100212	04/11/2007 Meeting Slides, "Overview of Technical Report 70, Revision 1."	Long, S. Williams, M.	Westinghouse	
4/20/2007	ML071360258	04/16-20/2007 Slide Presentation, "AP1000 Seismic Analyses Review."	Lindgren, D. A. Tunon-Sanjur, L.	Westinghouse	

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4/23/2007	ML070780321	Westinghouse AP1000 Combined License (COL) Pre-application Technical Report 33 - Request for Additional Information (TAC No. MD3191).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	TAC MD3191
4/23/2007	ML070860205	Westinghouse AP1000 Combined License (COL) Pre-application Technical Report 1, Construction Plan and Startup Schedule (APP-GW-GLR-036).	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	APP-GW-GLR-036
4/23/2007	ML071070099	03/22/2007 - 03/23/2007, Summary of Meeting with AP1000 and ESBWR Design-Centered Working Groups to Discuss Pre-Combined License Application Issues.	Kevern, T. A.	NRC/NRO/DNRL	
4/24/2007	ML071070567	Westinghouse AP1000 Technical Reports.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	AP1000
4/26/2007	ML071080405	Letter, Westinghouse, Re: Acceptance of Individual Related to Access to Safeguards Information	Matthews, D. B.	NRC/NRO/DNRL	
4/26/2007	ML071200220	AP1000 Piping Design Acceptance Criteria (DAC) Plan.	Sterdis, A.	Westinghouse	DCP/NRC1871
4/27/2007	ML070860414	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 6 - Request for Additional Information (TAC NO. MD2174).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	TAC MD2174
4/27/2007	ML071210088	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-006, Revision 2.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1873

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4/27/2007	ML071210096	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-052, Revision 0.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1872
4/30/2007	ML071170239	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 74a - Request for Additional Information (TAC NO. MD3838).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	TAC MD3838
4/30/2007	ML071290174	APP-GW-GLR-079, Rev. 0, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA." TR26	Keegan, C. P.	Westinghouse	DCP/NRC1877
4/30/2007	ML071290174	APP-GW-GLR-079, Rev. 0, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA." TR26	Keegan, C. P.	Westinghouse	DCP/NRC1877
5/1/2007	ML071270039	Second Request for Additional Information on Westinghouse AP1000 Document No. APP-GW-GLR-021, COL Information Items 5.3-1 and 5.3-4 (TAC No. MD2174).	Mitchell, M. A.	NRC/NRR/ADES/DCI/ CVIB	APP-GW-GLR-021 TAC MD2174
5/1/2007	ML071270039	Second Request for Additional Information on Westinghouse AP1000 Document No. APP-GW-GLR-021, COL Information Items 5.3-1 and 5.3-4 (TAC No. MD2174).	Mitchell, M. A.	NRC/NRR/ADES/ DCI/CVIB	APP-GW-GLR-021 TAC MD2174
5/2/2007	ML070871086	Westinghouse, RAI, Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 34 (TAC No. MD3648).	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	TAC MD3648

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5/4/2007	ML071170284	Westinghouse, RAI, Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 45. (TAC No. MD2495).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	DCP/NRC1753 TAC MD2495
5/4/2007	ML071290312	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-066, Revision 0. TR94	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-066, Rev 0 DCP/NRC1878
5/7/2007	ML071290170	AP1000 COL Standard Technical Report Submittal of Sensitive and Non-Sensitive APP-GW-GLR-079, Revision 0. TR26	Sterdis, A.	Westinghouse	DCP/NRC1877
5/7/2007	ML071290172	APP-GW-GLR-079-NS, Rev. 0, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA." Redacted Version. TR26		Westinghouse	DCP/NRC1877
5/8/2007	ML071080320	Change of Project Manager and Branch Assignment for AP1000 Design Certification Amendment - (Project Number 740).	Bergman, T. A.	NRC/NRO/DNRL	
5/8/2007	ML071080359	Change of Project Manager and Branch Assignment for AP1000 Design Certification Amendment - (Project Number 0740).	Bergman, T. A.	NRC/NRO/DNRL	
5/9/2007	ML071310068	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-055, Revision 0. TR11h	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1876

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5/10/2007	ML071270737	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 93 - Request for Additional Information (TAC No. MD4624).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	TAC MD4624
5/11/2007	ML071160237	04/11-12/2007 Summary of the Meeting to Discuss AP1000 Plant Operating Procedures.	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B1	
5/11/2007	ML071340345	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-113, Revision 0, "AP1000 Containment Vessel Shell: Material Specification." TR113	Sterdis, A.	Westinghouse	DCP/NRC1882
5/11/2007	ML071340349	Westinghouse AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-037, Revision 1, "AP1000 Test Specifications & Procedures." TR718	Sterdis, A.	Westinghouse	DCP/NRC1886
5/11/2007	ML071340353	Submittal of AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-101, Revision 0. TR101	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1885
5/11/2007	ML071350139	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-110, Revision 0. TR110	Sterdis, A.	Westinghouse	DCP/NRC1883

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5/14/2007	ML071200211	Acceptance for Review of Topical Report No. AP-TR-NS01, Containment Leak Rate Test Program Description, Revision 1 (Project No. 740; TAC MD5136).	Coffin, S. M.	NRC/NRO/DNRL/ AP1000B1	AP-TR-NS01 TAC MD5136
5/14/2007	ML071370085	AP1000 COL Standard Technical Report Submittal of Sensitive and Non-Sensitive APP-GW-GLN-022, Revision 1. TR97	Sterdis, A.	Westinghouse	DCP/NRC1869
5/14/2007	ML071370086	APP-GW-GLN-022-NS, Revision 1, "AP1000 Standard Combined License Technical Report DAS Platform Technology and Remote Indication Change." TR97	Cernuska, M. J.	Westinghouse	DCP/NRC1869
5/15/2007	ML071230370	Westinghouse, RAI, Westinghouse AP1000 Combined License Pre-Application Technical Report 3 - Request for Additional Information (TAC No. MD2358).	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B2	DCP/NRC1751 TAC MD2358
5/15/2007	ML071350512	SRM-SECY-07-0076 - Proposed Plan for Cooperation with China on the AP1000.	Vietti-Cook, A. L.	NRC/SECY	SECY-07-0076
5/15/2007	ML071360371	VR-SECY-07-0076, "Proposed Plan for Cooperation with China on the AP1000."	Jaczko, G. B. Klein,, D. E. Lyons, P. B. McGaffigan, E. Merrifield, J. S. Vietti-Cook, A. L.	NRC/Chairman NRC/OCM NRC/SECY	SECY-07-0076
5/16/2007	ML071060250	AP1000 Requests for Withholding Information from Public Disclosure (AW-07-2261).	Bloom, S. D.	NRC/NRO/DNRL/ AP1000B2	AW-07-2261

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5/16/2007	ML071370715	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-109, Revision 0.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1887
5/16/2007	ML071380080	Westinghouse Transmittal of Proprietary and Non-Proprietary Reports Related to Turbine Reliability.	Sterdis, A.	Westinghouse	DCP/NRC1889 WCAP-16650-P
5/17/2007	ML071370450	G20070319, Briefing Package Memo, Drop-In Visit by Westinghouse Officials, May 29-30, 2007.	McKenna, E. M.	NRC/NRO/DNRL/ AP1000B2	G20070319
5/17/2007	ML071410075	AP1000 COL Response to Request for Additional Information (TR #54).	Hutchings, D. F. Sterdis, A.	Westinghouse	APP-GW-GLR-033, Rev 0 DCP/NRC1890
5/17/2007	ML071410145	AP1000 COL Response to Request for Additional Information (TR #54).	Sterdis, A.	Westinghouse	DCP/NRC1891 TAC MD2551
5/17/2007	ML071410146	Enclosure 2, "Response to Request for Additional Information on Technical Report No. 54."		Westinghouse	DCP/NRC1891 RAI-TR54-021
5/18/2007	ML071290470	Westinghouse AP1000 Combined License Pre-Application Technical Report 9 - Request for Additional Information (TAC NO. MD1857).	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B2	TAC MD1847
5/18/2007	ML071410278	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-013, Revision 1.	Sterdis, A.	Westinghouse	AP1000 DCP/NRC1893
5/22/2007	ML071360008	04/16-20/2007 Summary of Open and Closed Meeting with Westinghouse to Discuss AP1000 Seismic Analyses.	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B1	

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10/26/2007	ML073030150	Submittal of APP-SSAR-GSC-642 Support Document for APP-GW-GLN-122 (TR 122).	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLN-122 APP-SSAR-GSC-642 AW-07-2350 DCP/NRC2030 TR 122
10/26/2007	ML073030175	APP-SSAR-GSC-642-NP, Rev. 0, "Assessment of Basis for Aerosol Plugging in AP1000 Containment Leak Paths for Radiological Design Bases Accidents."	Li, J.	Westinghouse	AP1000 DCP/NRC2030
10/26/2007	ML073120415	APP-SSAR-GSC-642, Rev. 0, "Assessment of Basis for Aerosol Plugging in AP1000 Containment Leak Paths for Radiological Design Bases Accidents."	Li, J.	Westinghouse	AP1000 DCP/NRC2030
10/26/2007	ML073120423	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-101 (TR 101) Revision 1.	Sterdis, A.	Westinghouse	DCP/NRC2029
10/26/2007	ML073120429	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-134 (TR 134), Revision 0.	Sterdis, A.	Westinghouse	APP-GW-GLR-134, Rev 0 DCP/NRC2031 TR 134

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10/26/2007	ML073090471	APP-GW-GLR-134, Revision 0, "AP1000 DCD Impacts to Support COLA Standardization."		Westinghouse	DCP/NRC2031
10/26/2007	ML073090473	APP-GW-GLR-134, Revision 0, "AP1000 DCD Impacts to Support COLA Standardization."		Westinghouse	DCP/NRC2031
11/2/2007	ML073110107	NRC Acceptance Review of AP1000 Design Certification Amendment Application.	Cummins, W. E.	Westinghouse	DCP/NRC2035
11/2/2007	ML073410071	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-006 (TR 62) Revision 3.	Sterdis, A.	Westinghouse	DCP/NRC2033
11/5/2007	ML073240104	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-143, Revision 0 (TR 143).	Sterdis, A.	Westinghouse	DCP/NRC2036
11/7/2007	ML073240108	AP1000 Standard Combined License Technical Report, APP-GW-GLN-105, Rev. 2, "Building and Structure Configuration, Layout, and General Arrangement Design Updates."		Westinghouse	
11/16/2007	ML073300299	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-020 Revision 4, (TR 05).	Sterdis, A.	Westinghouse	AP1000 DCP/NRC2043 TR 05
11/16/2007	ML073300302	Westinghouse AP1000 Response to NRC Regulatory Issue Summary 2007-08, Updated Licensing Submittal Information to Support the Design-Centered Licensing Review Approach - Updated Supplier Listing.	Sterdis, A.	Westinghouse	DCP/NRC2044

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11/21/2007	ML073300305	Transmittal of AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-045, Revision 1 (TR 57).	Sterdis, A.	Westinghouse	APP-GW-GLR-045, Rev 1 DCP/NRC2047 TR 57
11/21/2007	ML073400818	APP-GW-GLR-045-NS, Revision 1, "Nuclear Island: Evaluation of Critical Sections," Technical Report Number 57.		Westinghouse	DCP/NRC2047
11/21/2007	ML073410068	APP-GW-GLR-045, Revision 1, "Nuclear Island: Evaluation of Critical Sections," Technical Report Number 57.		Westinghouse	DCP/NRC2047
12/4/2007	ML073410069	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-006 (TR 62) Revision 4.	Sterdis, A.	Westinghouse	DCP/NRC2051
12/4/2007	ML073470914	AP1000 COL Standard Technical Report Submittal of APP-GQ-GLN-105, Revision 2 (TR 105).	Sterdis, A.	Westinghouse	DCP/NRC2048
12/4/2007	ML073470922	AP1000 Licensing Design Change Document, APP-GW-GLN-105, Rev 2, "Building and Structure Configuration, Layout, and General Arrangement Design Updates."		Westinghouse	
12/12/2007	ML073470925	AP1000 Design Specification Amendment Piping Licensing Proposal.	Cummins, W. E.	Westinghouse	DCP/NRC2059
12/12/2007	ML073480033	Transmittal of Proprietary and Non-Proprietary Summary Outlines to Support NRC Acceptance Review of Technical Report 74A.	Sterdis, A.	Westinghouse	DCP/NRC2058

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12/12/2007	ML073510258	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-144, Revision 0 (TR 144).	Sterdis, A.	Westinghouse	DCP/NRC2056 FOIA/PA-2009-0035
12/12/2007	ML073510265	AP1000 COL Standard Technical Report Submittal of APP-GW-134 (TR 134), Revision 1.	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-134, Rev 1 DCP/NRC2057 TR 134
12/13/2007	ML080100660	AP1000 Human Factors Engineering Program Plan Proposal.	Sterdis, A.	Westinghouse	APP-OCS-GBH-001 DCP/NRC2061
12/13/2007	ML073460098	AP1000 ASME Component Review Schedule.	Sterdis, A.	Westinghouse	DCP/NRC2060
12/20/2007	ML080150506	Westinghouse Presentation: Evaluation for High Frequency Seismic Input.	Lapay, W. S. Parello, J.	Westinghouse	
1/7/2008	ML080150513	Westinghouse AP1000 Combined License (COL) Pre-Application Technical Report 94 Rev. 1 - Request For Additional Information.	Miernicki, M. J.	NRC/NRO/DNRL/ AP1000B2	AP 1000 TAC MD5501
1/11/2008	ML080150526	AP1000 Technical Specifications I&C Bracketed Items Acceptance Issue.	Sterdis, A.	Westinghouse	AP1000 APP-GW-GLR-064, Rev 1 APP-GW-GSC-020 DCP/NRC2068 NUREG-1431, Rev 2
1/11/2008	ML080150529	AP1000 Piping DAC/Component COL Information Item 3.9-2 Acceptance Issue.	Sterdis, A.	Westinghouse	DCP/NRC2067
1/11/2008	ML080630100	Transmittal of Proprietary Information, AP1000 Containment Recirculation and IRWST Screen Design.	Sterdis, A.	Westinghouse	APP-GW-GLN-147, Rev 0 DCP/NRC2066
1/11/2008	ML080630101	APP-GW-GLN-147, Revision 0, "AP1000 Containment Recirculation and IRWST Screen Design."		Westinghouse	

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1/11/2008	ML080220387	APP-GW-GLN-147, Rev. 0, "AP1000 Recirculation and IRWST Screen Design."	McGinnis, C.	Westinghouse	
1/14/2008	ML080220389	AP1000 COL Information Items 3.6-1 and 3.9-2 Acceptance Issue Corrections.	Sterdis, A.	Westinghouse	DCP/NRC2070
1/14/2008	ML080220392	AP1000 COL Standard Technical Reports Submittal of APP-GW-GLR-134, Revision 3 (TR 134).	Sterdis, A.	Westinghouse	APP-GW-GLR-134, Rev 3 DCP/NRC2071 TR 134
1/14/2008	ML080430256	AP1000 Standard Combined License Technical Report, APP-GW-GLR-134, Revision 3, "AP1000 DCD Impacts to Support COLA Standardization."		Westinghouse	
1/14/2008	ML080370179	AP1000 Standard Combined License Technical Report, APP-GW-GLR-134, Revision 2, "AP1000 DCD Impacts to Support COLA Standardization."		Westinghouse	M
2/5/2008	ML080510189	02/05/08 - Email subject: Schedule for Technical Specification Submittal.	Sterdis, A.	Westinghouse	TR 74
2/15/2008	ML080600452	Letter to Westinghouse: Review Schedule for the AP1000 Design Certification Amendment Application Revision 16.	Gleaves, W. C.	NRC/NRO/DNRL/ AP1000B2	
2/15/2008	ML080600520	10 CFR 50.46 Report for the AP1000 Standard Plant Design.	Cummins, W. E.	Westinghouse	DCP/NRC2074

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2/27/2008	ML080220403	G20080147/EDATS: OEDO-2008-0166 - W.E. Cummins Ltr re Schedule for AP1000 Design Certification Amendment	Cummins, W. E.	Westinghouse	DCP/NRC2089 G20080147 OEDO-2008-0166
2/27/2008	ML080630099	Proposed Combined Operating License Information Item for Cyber Security.	Lindgren, D. A.	Westinghouse	APP-GW-GLR-104 DCP/NRC2086
2/28/2008	ML091470066	Piping Design Acceptance Criteria (DAC).	Matthews, D. B.	NRC/NRO/DNRL	
2/28/2008	ML080650717	Correction to AP1000 Submittal of APP-GW-GLN-147, Revision 0.	Lindgren, D. A.	Westinghouse	DCP/NRC2066
2/28/2008	ML080710111	APP-OCS-GGR-110-P, Rev. 1, "AP1000 Technical Support Center and Emergency Operations Facility Workshop."	Reed, J.	Westinghouse	DCP/NRC2473
3/3/2008	ML080710112	Submittal of Proprietary and Non-Proprietary Versions of WCAP-16914, Evaluation of Debris Loading Head Loss Tests for AP1000 Recirculation Screens and In-Containment Refueling Water Storage Tank Screens.	Lindgren, D. A.	Westinghouse	AW-08-2394 DCP/NRC2094 WCAP-16914
3/5/2008	ML080710113	AP1000 NRC Program Status Presentation - I&C and Human Factors Engineering February 5, 2008 - February 8, 2008 Submittal of Proprietary and Non-Proprietary Presentations.	Lindgren, D. A.	Westinghouse	AW-08-2391 DCP/NRC2090
3/5/2008	ML080710114	APP-GW-GLY-003, Revision 0, "AP1000 I&C Meeting Licensing Overview."	Sterdis, A.	Westinghouse	DCP/NRC2090

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3/5/2008	ML080701114	APP-PMS-GLY-003-NP, Revision 0, "NRC Program Status Presentation I&C and Human Factors Engineering."		Westinghouse	DCP/NRC2090
3/5/2008	ML080710318	APP-PMS-GLY-003, Revision 0, "NRC Program Status Presentation I&C and Human Factors Engineering."		Westinghouse	DCP/NRC2090
3/6/2008	ML080730026	Submittal of APP-GW-GLR-064, Revision 2, Technical Report 74A.	Lindgren, D. A.	Westinghouse	AP1000 APP-GW-GLR-064, Rev 2 DCP/NRC2098
3/7/2008	ML080800166	Westinghouse Submittal of APP-GW-GLE-001, Revision 0, "Impact of Annex Building Expansion and Condenser Air Removal Stack Location on the Control Room Atmospheric Dispersion Factors".	Lindgren, D. A.	Westinghouse	AP1000 DCP/NRC2099
3/14/2008	ML080800117	Submittal of Certain Supporting Information to the AP1000 Docket.	Gleaves, W. C.	NRC/NRO/DNRL/ AP1000B2	AP1000
3/17/2008	ML080840188	Submittal of Proprietary and Non-Proprietary Versions of APP-OCS-GGR-110-P, Revision 1, "AP1000 Technical Support Center & Emergency Operations Facility Workshop."	Sisk, R. B.	Westinghouse	APP-OCS-GGR-110-P, Rev 1 DCP/NRC2101
3/18/2008	ML080850404	AP1000 DCD Impact Document Submittal of APP-GW-GLE-003, Revision 0, "Addition of Site Specific System Designator YFS and FPS Change for Administration Building."	Sisk, R. B.	Westinghouse	APP-GW-GLE-003, Rev 0 DCP/NRC2105

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3/20/2008	ML080920454	Westinghouse Transmittal of Revision 0 of APP-GW-GLE-004, "Soil and Seismic Parameter Change," Identifying Changes to the AP1000 Design Control Document.	Sisk, R. B.	Westinghouse	DCP/NRC2106
3/20/2008	ML080920487	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-134, Revision 4 (TR 134).	Sisk, R. B.	Westinghouse	APP-GW-GLR-134, Rev 4 DCP/NRC2107 TR 134
3/28/2008	ML080920489	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-079 (TR26), Revision 3.	Sisk, R. B.	Westinghouse	APP-GW-GLR-079, Rev 3 AW-08-2404 DCP/NRC2108 TR 26
3/28/2008	ML080980257	AP1000 DCD Impact Document Submittal of APP-GW-GLE-002, Revision 0.	Sisk, R. B.	Westinghouse	APP-GW-GLE-002, Rev 0 DCP/NRC2111 GSI-191
3/28/2008	ML080940161	APP-GW-GLE-002, Revision 0, "Impacts to AP1000 DCD to Address Generic Safety Issue (GSI)-191."		Westinghouse	DCP/NRC2111 GSI-191
4/3/2008	ML081020229	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-126, Revision 0 (TR 126).	Sisk, R. B.	Westinghouse	APP-GW-GLR-126, Rev 0 DCP/NRC2109 TR 126
4/7/2008	ML081000080	Submittal Of Certain Supporting Information On The AP1000 Docket. NRC Requests that Westinghouse Submit APP-OCS-J1R-110 on the Docket.	Jaffe, D. H.	NRC/NRO/DNRL/ AP1000B2	APP-OCS-J1R-110, Rev. 0
4/9/2008	ML081220429	Submittal of Matrix that Provides an Assessment of the AP1000 Design with Respect to Regulatory Guide 1.82, Revision 3.	Cummins, W. E.	Westinghouse	DCP/NRC2116 RG-1.082, Rev 3

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4/17/2008	ML081410759	Submittal of Certain Supporting Information to The AP1000 Docket.	Jaffe, D. H.	NRC/NRO/DNRL/ AP1000B2	
4/28/2008	ML081370147	Submittal of APP-PXS-GLR-001, Revision 0, "Impact on AP1000 Post-LOCA Long Term Cooling of Postulated Containment Sump Debris."	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2127
5/16/2008	ML081430068	AP1000 DCD Impact Document Submittal of APP-GW-GLE-006, Revision 0, "ASME Code Citation."	Sisk, R. B.	Westinghouse	DCP/NRC2136
5/20/2008	ML081440047	Meeting Handouts - Modular Construction and ITAAC.	Lindgren, D.	Westinghouse	APP-GW-GLY-004
5/20/2008	ML081620100	Review Schedule for the AP1000 Design Certification Amendment Application, Revision 16.	Sisk, R. B.	Westinghouse	DCP/NRC2133
5/20/2008	ML081550223	AP1000 DCD Impact Document Submittal of APP-GW-GLE-009, Revision 0.	Sisk, R. B.	Westinghouse	APP-GW-GLE-009, Rev 0 DCP/NRC2137
5/22/2008	ML081580162	NuStart AP1000 COL Project, AP1000 Design Certificate Amendment, Payment of NRC Invoices.	Cummins, W. E. Kray, M. C.	NuStart Energy Development, LLC Westinghouse	DCP/NRC2155
5/30/2008	ML081700166	AP1000 Submittal of APP-RXS-Z0R-001, Revision 1.	Sisk, R. B.	Westinghouse	APP-RXS-Z0R-001, Revision 1 DCP/NRC2145
6/3/2008	ML082390111	Methodologies Used for the AP1000 Evaluation of Long-Term Core Cooling.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2140
6/13/2008	ML081490403	AP1000 DCD Impact Document Submittal of APP-GW-GLE-007, Revision 0.	Sisk, R. B.	Westinghouse	AP1000 APP-GW-GLE-007, Rev 0 DCP/NRC2159

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6/20/2008	ML081830145	Westinghouse Submittal of List of AP1000 Activities for Review During NRC QA Audit in October 2008.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2168
6/27/2008	ML081830146	06/27/2008 Review Schedule For AP1000 Revision 16.	Bergman, T. A.	NRC/NRO/DNRL	AP1000
6/27/2008	ML081830156	AP1000 DCD Impact Document Submittal of APP-GW-GLE-012, Revision 0.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2184
6/27/2008	ML081850468	AP1000 DCD Impact Document Submittal of APP-GW-GLE-036, Revision 0.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2185
6/27/2008	ML081850575	AP1000 DCD Impact Document Submittal of APP-GW-GLE-002, Revision 1.	Sisk, R. B.	Westinghouse	AP1000 APP-GW-GLE-002, Rev 1 DCP/NRC2181
6/27/2008	ML081850576	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-134, Revision 5 (TR 134).	Sisk, R. B.	Westinghouse	APP-GW-GLR-134, Rev 05 DCP/NRC2183 TR 134
6/27/2008	ML081850577	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-104, (TR #104).	Sisk, R. B.	Westinghouse	AP1000 APP-GW-GLR-104 DCP/NRC2186 WCAP-16791-NP Rev. 1 WCAP-16791-P Rev. 1
6/27/2008	ML081850550	WCAP-16791-NP, Rev. 1, "AP 1000 Cyber Security Implementation."		Westinghouse	AP1000 DCP/NRC2186
6/27/2008	ML081850551	WCAP-16791-P, Rev. 1, "AP 1000 Cyber Security Implementation."	Gasparovic, M. A.	Westinghouse	AP1000 DCP/NRC2186

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6/30/2008	ML081920164	APP-GW-GLR-134, Revision 05, AP1000 Standard Combined License Technical Report AP1000 DCD Impacts to Support COLA Standardization.		Westinghouse	DCP/NRC2183 TR 134
6/30/2008	ML081920165	APP-GW-GLR-134, Rev 5, "AP1000 Standard Combined License Technical Report AP1000 DCD Impacts to Support COLA Standardization."		Westinghouse	DCP/NRC2183 TR 134
6/30/2008	ML081900574	APP-GW-GLE-016-NS, Revision 0, "AP1000 DCD Impact Document, Impact of In-core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package (IHP)."		Westinghouse	DCP/NRC2190
6/30/2008	ML081900575	APP-GW-GLE-016, Revision 0, "AP1000 DCD Impact Document Impact of In-Core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package (IHP)."		Westinghouse	DCP/NRC2190
7/1/2008	ML081900576	Westinghouse - AP1000 DCD Impact Document Submittal of APP-GW-GLR-045, Revision 2 (TR-57).	Sisk, R. B.	Westinghouse	APP-GW-GLR-045, Rev 2 DCP/NRC2188
7/1/2008	ML081920163	APP-GW-GLR-045 NS, Revision 2, "Nuclear Island: Evaluation of Critical Sections" (Public Version).		Westinghouse	DCP/NRC2188
7/1/2008	ML082000076	APP-GW-GLR-045, Revision 2, "Nuclear Island: Evaluation of Critical Sections" (Sensitive Version).		Westinghouse	DCP/NRC2188

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7/7/2008	ML082210170	AP1000 DCD Impact Document Submittal of APP-GW-GLE-016, Revision 0.	Sisk, R. B.	Westinghouse	APP-GW-GLE-016, Rev 0 DCP/NRC2190
7/15/2008	ML082250353	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-026, Revision 1 (TR 44).	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2202 TR 44
8/6/2008	ML082250358	AP1000 DCD Impact Document Submittal of APP-GW-GLE-034, Revision 0.	Sisk, R. B.	Westinghouse	DCP/NRC2222
8/7/2008	ML082250359	Submittal of Proprietary and Non-Proprietary of APP-FA01-T2R-001, Evaluation of Debris Loading Head Loss Tests for AP1000 Simulated Fuel Assembly During Post-Accident Recirculation.	Sisk, R. B.	Westinghouse	APP-FA01-T2R-001 AW-08-2466 DCP/NRC2227
8/7/2008	ML082350194	APP-FA01-T2R-001-NP, Revision 0, "Evaluation of Debris Loading Head Loss Tests for AP1000 Simulated Fuel Assembly During Post-Accident Recirculation."		Westinghouse	DCP/NRC2227
8/7/2008	ML082350196	APP-FA01-T2R-001, Revision 0, "Evaluation of Debris Loading Head Loss Tests for AP1000 Simulated Fuel Assembly During Post-Accident Recirculation."		Westinghouse	DCP/NRC2227
8/7/2008	ML082350197	APP-MY03-GLY-001-NP, Revision 0, "Presentations from the April 16th 2008 Meeting Regarding AP 1000 Sump Screen Design."	Andreychek, T. S. Monahan, J. S.	Westinghouse	DCP/NRC2149

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8/7/2008	ML082350193	APP-MY03-GLY-001, Revision 0, "Presentations from the April 16th 2008 Meeting Regarding AP 1000 Sump Screen Design," Part 2 of 3.		Westinghouse	DCP/NRC2149
8/7/2008	ML082350195	APP-MY03-GLY-001, Revision 0, "Presentations from the April 16th 2008 Meeting Regarding AP1000 Sump Screen Design." Part 3 of 3.		Westinghouse	DCP/NRC2149
8/13/2008	ML082330095	Submittal of Proprietary and Non-Proprietary Presentations from the April 16th 2008 Meeting Regarding AP1000 Sump Screen Design.	Sisk, R. B.	Westinghouse	AW-08-2444 DCP/NRC2149
8/13/2008	ML091660340	APP-MY03-GLY-001, Revision 0, "Presentations from the April 16th 2008 Meeting Regarding AP 1000 Sump Screen Design." Part 1 of 3.		Westinghouse	DCP/NRC2149
8/15/2008	ML082380866	Westinghouse Submittal of Proposed Changes to AP1000 Design Control Document to Remove Piping Design Acceptance Criteria and Revise COL Information Item.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2228
8/19/2008	ML082660365	Enclosure 5 - Westinghouse Proprietary, Fuel Rack Drawing Package, APP-FS02-V1-001, Revision 1, 8/19/08.		Westinghouse	DCP_NRC_002518

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8/21/2008	ML083220482	Westinghouse Request to Include Control Document Amendment, Revision 17 in the Assessment of the NRC Review Schedule Before Reissuing the Schedule for AP1000 While Westinghouse is Submitting an Update to Rev. 16 in Mid-September.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2232
9/17/2008	ML083220503	AP1000 Design Control Document REV17 Submittal Overview.		Westinghouse	AP1000
9/22/2008	ML083220506	Update to Westinghouse's Application to Amend the AP1000 Design Certification Rule.	Sisk, R. B.	Westinghouse	AP1000 APP-GW-GL-700, Rev 17 DCP/NRC2266
9/22/2008	ML083220507	Westinghouse AP1000 Cover Letter Rev. 17 - Oath and Affirmation Letter	Sisk, R. B.	Westinghouse	APP-GW-GL-700.CVR.S APP-GW-GL-700.CVR.S.17 WESTINGHOUSE
9/22/2008	ML083220508	Westinghouse AP1000 Design Control Document Rev. 17 - Design Control Document Introduction	Sisk, R. B.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.17 WESTINGHOUSE
9/22/2008	ML083220509	Westinghouse AP1000 Design Control Document Rev. 17 - Tier 1 - Change Roadmap	Sisk, R. B.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.17 WESTINGHOUSE
9/22/2008	ML083230167	Westinghouse AP1000 Design Control Document Rev. 17 - Tier 1 - List of Effective Pages	Sisk, R. B.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.17 WESTINGHOUSE
9/22/2008	ML083120490	Westinghouse AP1000 Design Control Document Rev. 17 - Tier 1 - Table of Contents	Sisk, R. B.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.17 WESTINGHOUSE
9/22/2008	ML083120491	Westinghouse AP1000 Cover Letter Rev. 17 - Oath and Affirmation Letter	Sisk, R. B.	Westinghouse	APP-GW-GL-700.CVR.P APP-GW-GL-700.CVR.P.17 WESTINGHOUSE
9/29/2008	ML083120287	Westinghouse AP1000 Revision 17, DCD Change Matrix.		Westinghouse	DCP/NRC2267

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9/29/2008	ML082940062	Transmittal of One (1) Copy of the Proprietary Version of the "Reviewer's Aide and Change Matrix for APP-GW-GI-700, AP1000 Design Control Document, Revision 17."	Sisk, R. B.	Westinghouse	APP-GW-GL-700 DCP/NRC2267
10/14/2008	ML083220483	Submittal of Revision 17 Errata Disks.	Sisk, R. B.	Westinghouse	DCP/NRC2275
10/16/2008	ML083090461	Westinghouse, Submittal of Revision 1, APP-GW-GLR-115, "Effect of High Frequency Seismic Content on SSCs," Technical Report 115 (TR-1 15) to Provide an Evaluation of Effects of High Frequency Seismic Input on AP1000 Design.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2276
10/24/2008	ML083290463	DCD Revision 17.	Sisk, R. B.	Westinghouse	DCP/NRC2281
11/3/2008	ML083330290	11/03/2008 AP1000 Dose Analysis Meeting Slides.		Westinghouse	
11/20/2008	ML083450729	Westinghouse AP1000 Updated Piping Review Schedule.	Sisk, R. B.	Westinghouse	DCP/NRC2293
11/25/2008	ML083590250	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-066, Revision 2.	Winters, J.	Westinghouse	APP-GW-GLR-066, Rev 2 DCP/NRC2297
12/15/2008	ML083470258	ACC Slide Presentation		Westinghouse	
12/17/2008	ML083500308	Transmittal of Westinghouse Quality Assurance Documents.	Sisk, R. B.	Westinghouse	DCP/NRC2327
12/30/2008	ML083520635	Safety Evaluation Of The Westinghouse AP1000 Generic Pressure-Temperature Limits Report (PTLR) - APP-RXS-Z0R-001.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	APP-RXS-Z0R-001, Revision 1

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12/30/2008	ML090220273	Summary of the October 20 - 24, 2008, On-Site Review of AP1000 Piping and Support Design.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	
12/30/2008	ML090220274	10/13-17/2008 - Summary of On-Site Review of the AP1000 Component Design.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	AP1000
1/19/2009	ML090300139	Westinghouse Submittal of Proposed Changes to AP1000 Design Control Document to Remove the Piping Design Acceptance Criteria.	Sisk, R. B.	Westinghouse	DCP/NRC2350
1/19/2009	ML090410368	Westinghouse - AP1000 Onsite Chemical RAI Revisions.	Sisk, R. B.	Westinghouse	DCP/NRC2345
1/28/2009	ML090410369	AP1000 ASME Code Citation Change.	Sisk, R. B.	Westinghouse	DCP/NRC2364
1/31/2009	ML090410367	APP-GW-GLE-NP, Rev 1, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000."		Westinghouse	DCP/NRC2368
1/31/2009	ML090500732	APP-GW-GLE-026, Rev 1, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis for AP1000."		Westinghouse	DCP/NRC2368
2/3/2009	ML090680308	AP1000 DCD Impact Document Submittal of APP-GW-GLE-026, Revision 1.	Sisk, R. B.	Westinghouse	APP-GW-GLE-026, Rev 1 AW-09-2525 DCP/NRC2368
2/3/2009	ML090480601	Westinghouse Response to NRC Inspection Report No. 05200006/2008-201, Notice of Nonconformance.	Sisk, R. B.	Westinghouse	DCP/NRC2371 IR-08-201 QLA/NRC0001

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2/5/2009	ML090480602	02/05-06/2009 Meeting Presentation on "Westinghouse Fuel Performance Update Meeting".		Westinghouse	AW-09-2526 LTR-NRC-09-7
2/11/2009	ML090480603	Submittal of Proprietary and Non-Proprietary of APP-MY03-T2C-002, Evaluation of Debris Loading Head Loss Tests Across AP1000 Fuel Assemblies During Post-Accident Recirculation.	Sisk, R. B.	Westinghouse	APP-MY03-T2C-002 AW-09-2528 DCP/NRC2376
2/11/2009	ML090490714	WCAP-17028-NP, Revision 0, "Evaluation of Debris Loading Head Loss Experiments Across AP1000 Fuel Assemblies During Post-Accident Recirculation."	McNamee, K. F. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002 DCP/NRC2376
2/11/2009	ML090540843	WCAP-17028-P, Revision 0, "Evaluation of Debris Loading Head Loss Experiments Across AP1000 Fuel Assemblies During Post-Accident Recirculation."	McNamee, K. F. Schulz, T. L. Song, .Y J.	Westinghouse	APP-MY03-T2C-002 DCP/NRC2376
2/12/2009	ML090060323	Presentation by Westinghouse, AP1000 Main Control Room In-leakage.		Westinghouse	
2/12/2009	ML090490094	Presentation by Westinghouse, AP1000 Post LOCA Recirculation Screens and Downstream Effects.		Westinghouse	
2/13/2009	ML090290160	Safety Evaluation Report with Open Items for Chapter 14 of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	NUREG-1793 S2

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2/13/2009	ML090370941	Westinghouse, Submittal of Annual Report to Document any Emergency Core Cooling System Evaluation Model Changes or Errors for AP1000 Standard Plant Design.	Sisk, R. B.	Westinghouse	DCP/NRC2373
2/17/2009	ML090500209	Letter to Westinghouse NRC Review of Selection 6 of the AP1000 Design Control Document.	Gleaves, W. C.	NRC/NRO/DNRL	
2/20/2009	ML090560606	Request for Additional Information Related to AP1000 Physical Security.	Buckberg, P. H.	NRC/NRO/DNRL	
2/20/2009	ML090790512	Requests for Additional Information Related to AP1000 Standard Combined License Technical Report 94, APP-GW-GLR-066, "AP1000 Safeguards Assessment".	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	AP1000 APP-GW-GLR-066
2/27/2009	ML091670372	NRC Review of Section 3.2, 3.6.2, And Piping Design Acceptance Criteria of the Design Control Document.	Gleaves, B. C.	NRC/NRO/DNRL	
3/13/2009	ML090840305	Submittal of Proprietary and Non-Proprietary Versions of APP-GW-JJ-002-P, Revision 2, "FMEA of AP1000 Protection & Safety Monitoring System."	Sisk, R. B.	Westinghouse	APP-GW-JJ-002-P, Rev 2 DCP/NRC2395
3/13/2009	ML090650149	Westinghouse AP1000 ASME Code Cases.	Sisk, R. B.	Westinghouse	DCP/NRC2402
3/16/2009	ML091130163	03/16/2009 NRC MCR VES Design Power Point Presentation.		Westinghouse	
3/24/2009	ML083540142	Chapter 7 of AP1000 Design Certification Amendment Review.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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3/27/2009	ML090970336	AP1000 ITAAC Update - Overview of AP1000 Status Update and Work with NRC CIP Taskforce Meeting.	Ray, T.	Westinghouse	
4/1/2009	ML090770458	Letter - Safety Evaluation Report with Open Items for Chapter 17 of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	Matthes, W. D.	NRC/NRO/DNRL	NUREG-1793 S2
4/2/2009	ML090970254	Westinghouse Submittal of APP-RXS-Z0R-001, Revision 2, "AP1000 Generic Pressure Temperature Limits Report," Response to Request for Additional Information in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP/NRC2416
4/3/2009	ML090370180	Revision To Review Schedule For AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
4/3/2009	ML091040743	Westinghouse Submittal of Shield Building Schedule.	Sisk, R. B.	Westinghouse	DCP/NRC2421
4/7/2009	ML091040771	Propriety Information Review - Safety Evaluation Report With Open Items For Chapter 8, "Electrical Power System," of NUREG-1793, Supplement 2-AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
4/9/2009	ML090920464	Pipe Break Hazard Evaluation.		Westinghouse	
4/9/2009	ML091340740	NRC Break Hazard Presentation.		Westinghouse	
4/14/2009	ML091340748	03/20/2009 Westinghouse Meeting Presentation on AP1000 Long-Term Cooling Debris Issues Resolution.	Schulz, T. L.	Westinghouse	

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4/17/2009	ML091120718	AP1000 Safety-Related Equipment Seismic Qualification Testing Priority 4 Item 4.	Parello, J.	Westinghouse	
4/17/2009	ML091280024	Damping Value for Seismic Qualification Analysis of AP1000 Electrical Cabinets and Panels Priority 4 Item 4.	Parello, J.	Westinghouse	
4/20/2009	ML091180432	Westinghouse Draft Chapters 8 and 17 SER Review and Determination of no Proprietary Information Included in the SER.	Sisk, R. B.	Westinghouse	DCP/NRC2434
4/21/2009	ML091280023	Response to NRC Letter 03/24/2009 Regarding Chapter 7 of AP1000 Design Certification Amendment Review.	Sisk, R. B.	Westinghouse	DCP/NRC2440
4/23/2009	ML091190299	Schedule for Action Item Follow Up to NRC Review Meeting on Seismic Issues.	Sisk, R. B.	Westinghouse	DCP/NRC2444
4/23/2009	ML091190713	Westinghouse Acknowledgement of Receipt of Revision to Review Schedule for AP1000 Design Certification Amendment.	Sisk, R. B.	Westinghouse	DCP/NRC2439
4/24/2009	ML091200580	Westinghouse Submittal of Responses to Action Item 10 from the March 18 and 19, 2009 Meeting Regarding AP1000 Shield Building Design.	Sisk, R. B.	Westinghouse	DCP/NRC2448
4/27/2009	ML091470061	Westinghouse - AP1000 Writer's Guideline Documents.	Sisk, R. B.	Westinghouse	DCP/NRC2450
4/28/2009	ML091470065	Westinghouse AP1000 ASME Code Case N-782.	Sisk, R. B.	Westinghouse	DCP/NRC2452

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4/30/2009	ML091470067	APP-OCS-GEH-420, Rev. B, "AP1000 Human Factors Engineering Discrepancy Resolution Process."	Van Meter, T. W.	Westinghouse	DCP/NRC2473
4/30/2009	ML091470069	APP-OCS-GBH-001, Rev. 1, "AP1000 Human Factors Engineering Program Plan."	Reed, J. I.	Westinghouse	DCP/NRC2473
4/30/2009	ML091470070	APP-OCS-J1R-210, Rev. 1, "AP1000 Operational Sequence Analysis 2 (OSA-2) Implementation Plan."	Shaffer, M. C.	Westinghouse	DCP/NRC2473
4/30/2009	ML091470071	WNA-DS-01046-GEN, Rev. 2, "RRAS Distributed Control & Information Systems, Standard Computerized Procedures System Functional Requirements."	Dudics, R. A.	Westinghouse	DCP/NRC2473
4/30/2009	ML091470073	APP-OCS-GEH-120, Rev. B, "AP1000 Human Factors Engineering Design Verification Plan."	Van Meter, T. W.	Westinghouse	DCP/NRC2473
4/30/2009	ML091120620	APP-OCS-GEH-220, Rev. B, "AP1000 Human Factors Engineering Task Support Verification Plan."	Van Meter, T. W.	Westinghouse	DCP/NRC2473
4/30/2009	ML091120769	APP-OCS-J1-020, Rev. 1, "Computerized Procedures System Functional Requirements."	Dudics, R. A.	Westinghouse	DCP/NRC2473
5/4/2009	ML091460242	Submittal of Proprietary Responses to Action Item 9 from the March 18 and 19, 2009 Meeting Regarding AP1000 Shield Building Design.	Sisk, R. B.	Westinghouse	AW-09-2570 DCP/NRC2456
5/12/2009	ML091410433	Westinghouse Draft Chapter 4 SER Review.	Sisk, R. B.	Westinghouse	DCP/NRC2488

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5/14/2009	ML091420104	Submittal of Proprietary and Non-Proprietary Documents for Docketing (RAI-SRP18-COLP708).	Sisk, R. B.	Westinghouse	AW-09-2577 DCP/NRC2473 RAI-SRP18-COLP-08
5/15/2009	ML091420105	AP1000 DCD Impact Document Submittal of APP-GW-GLE-002, Revision 2.	Sisk, R. B.	Westinghouse	DCP/NRC2480 GSI-191
5/15/2009	ML091420106	Submittal of Proprietary and Non-Proprietary Versions of APP-GW-GLN-147 (TR 147), Revision 2.	Sisk, R. B.	Westinghouse	APP-GW-GLN-147, Rev 2 AW-09-2580 DCP/NRC2482 TR 147
5/15/2009	ML091470691	APP-GW-GLE-147-NP, Rev. 2, "AP 1000 Containment Recirculation and IRWST Screen Design."		Westinghouse	DCP/NRC2482
5/15/2009	ML091470696	APP-GW-GLN-147, Rev. 2, "AP1000 Containment Recirculation and IRWST Screen Design," Technical Report Number 147.		Westinghouse	DCP/NRC2482
5/20/2009	ML091470707	05/20/2009 Meeting Slides, AP1000 Design Certification Pipe Rupture Hazard Evaluation Meeting with NRC."		Westinghouse	
5/20/2009	ML091470074	05/20/2009 Meeting Slides, AP1000 As-Designed Pipe Rupture Hazards Analysis (PRHA).		Westinghouse	
5/20/2009	ML091480387	05/20/2009 Meeting Slides, Pipe Rupture Hazard Evaluation Design Certification Review Licensing Background and Approach.		Westinghouse	

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5/22/2009	ML091480388	Submittal of Responses to Action Item 12 from the March 18 and 19, 2009 Meeting Regarding AP1000 Shield Building Design.	Sisk, R. B.	Westinghouse	AP1000 DCP/NRC2490
5/22/2009	ML091480015	Submittal of Proprietary Responses to Action Items from the March 18 and 19, 2009 Meeting Regarding AP1000 Shield Building Design.	Sisk, R. B.	Westinghouse	AW-09-2587 DCP/NRC2489
5/26/2009	ML091340725	AP1000 Response to Request for Additional Information in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP/NRC2497
5/28/2009	ML091340774	Seismic Incoherent SSI Methodology Applied to AP1000 NPP at Bellefonte Site Priority 1 Item 1.	Ghiocel, D. M.	Westinghouse	
5/28/2009	ML091420474	Adjacent Building Seismic Design Priority 4 Item 3.		Westinghouse	
5/28/2009	ML091480740	SCV Equivalent Static Analysis Priority 2 Item 3.	Schoonmaker, B.	Westinghouse	
5/28/2009	ML091520089	Effects of High Frequency on Piping.		Westinghouse	
5/28/2009	ML091530125	Proprietary - Adjacent Building Seismic Design Priority 4 Item 3.		Westinghouse	
5/28/2009	ML091530128	Westinghouse Draft Chapter 19 SER Review.	Sisk, R. B.	Westinghouse	DCP/NRC2498
5/29/2009	ML091620255	AP1000 DCD Impact Document Submittal of APP-PXS-GLR-001, Rev. 1, "Impact on AP1000 Post-LOCA Long Term Cooling of Postulated Containment Sump Debris."	Sisk, R. B.	Westinghouse	APP-PXS-GLR-001, Rev 1 DCP/NRC2494

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5/29/2009	ML091620256	Submittal of Proprietary Document APP-OCS-GEH-320, Revision B "AP1000 Human Factors Engineering Integrated System Validation Plan" for Docketing.	Sisk, R. B.	Westinghouse	APP-OCS-GEH-320, Rev B AW-09-2592 DCP/NRC2509
5/31/2009	ML091520424	APP-GW-GLR-065-NP, Revision 1, "AP1000 I&C Data Communication and Manual Control of Safety Systems and Components."	Crew, A. W.	Westinghouse	DCP/NRC2513
5/31/2009	ML090700697	APP-GW-GLR-065-P, Revision 1, "AP1000 I&C Data Communication and Manual Control of Safety Systems and Components."	Crew, A. W.	Westinghouse	DCP/NRC2513
6/1/2009	ML091470625	Safety Evaluation Report Related to the Review of AP1000 Standard Combined License Technical Report Number 30, Revision 0.	Terao, D. A.	NRC/NRO/DE/CIB1	
6/2/2009	ML091530262	Enclosure - Safety Evaluation Report Chapter 16. Technical Specifications.		NRC/NRO/DNRL	
6/2/2009	ML091660338	05/20/2009 Summary of Public Meeting to Discussed Pipe Break Hazard Planning With Westinghouse on the AP1000 Design Certification Amendment.	Gleaves, B. C.	NRC/NRO/DNRL	
6/2/2009	ML091390631	06/15/09 Notice of Meeting with Westinghouse on AP1000 Design Certification Amendment Structural And Seismic Issues.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	

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6/4/2009	ML091600141	ACRS Meeting with the U.S. Nuclear Regulatory Commission, - June 4, 2009, Slides.	Bonaca, M. V.	NRC/ACRS	
6/4/2009	ML093170673	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-005, Revision 3 (TR-09).	Sisk, R. B.	Westinghouse	APP-GW-GLR-005, Rev 3 DCP/NRC2515
6/4/2009	ML091610613	Westinghouse, LLC - Submittal of Proprietary and Non-Proprietary Versions of APP-GW-GLR-071, Revision 2.	Sisk, R. B.	Westinghouse	
6/8/2009	ML091620251	Summary of Audit of AP1000 Reactor Coolant Pump External Heat Exchanger Design Report.	Donoghue, J. E.	NRC/NRO/DSRA/ SRSB	
6/8/2009	ML091620252	Supplemental Information Regarding AP1000 ASME Code Case N-782.	Sisk, R. B.	Westinghouse	DCP/NRC2519 RG-1.084, Rev 34
6/8/2009	ML091620253	Submittal of Proprietary Information on Structural Model Node Locations.	Sisk, R. B.	Westinghouse	AP1000 AW-09-2598 DCP/NRC2524
6/8/2009	ML091660336	Submittal of Proprietary and Non-Proprietary Versions of APP-GW-GLR-065, Revision 1.	Sisk, R. B.	Westinghouse	APP-GW-GLR-065, Rev 1 DCP/NRC2513
6/9/2009	ML091660342	06/15/09 - 06/16/09 Notice of Public Meeting with Westinghouse on AP1000 Design Certification Amendment Structural Review Of Shield Building.	Gleaves, B. C.	NRC/NRO/DNRL	
6/9/2009	ML091670169	Enclosure 9 - Westinghouse Proprietary Roadmap of Draft NRC Audit Items and Coverage in Criticality Analysis.		Westinghouse	DCP_NRC_002518

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6/10/2009	ML091670170	Draft Chapter 16 SER Review.	Sisk, R. B.	Westinghouse	DCP_NRC002525
6/12/2009	ML091800405	Westinghouse - Proprietary Review of Draft Chapter 18 SER Review - DCD Revision 16.	Sisk, R. B.	Westinghouse	DCP_NRC_002529
6/15/2009	ML090860852	Meeting Handout: AP1000 Shield Building Steel Plate Modular Design Regulatory Issues by D. Lindgren, Westinghouse.	Lindgren, D. A.	Westinghouse	
6/16/2009	ML091190346	Draft Safety Evaluation for AP1000 Design Certification Document Chapter 15, Revision 17.	Donoghue, J. E.	NRC/NRO/DSRA/ SRSB	
6/17/2009	ML091700120	Enclosure - Safety Evaluation Report Reactor Coolant System and Connected Systems Chapter 5.		NRC/NRO/DNRL/ NWE2	NUREG-1793
6/17/2009	ML091410535	BLN QA Program Design, Chapter 17 from ISL Chapter Day Comments 5/29/09.		- No Known Affiliation	
6/17/2009	ML091520136	Westinghouse Submittal of Matrix for Assessment of AP1000 Design With Respect to Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."	Sisk, R. B.	Westinghouse	DCP/NRC-002535 RG-1.082, Rev. 3
6/18/2009	ML091540494	Letter - Safety Evaluation Report with Open Items for Chapter 11, "Radioactive Waste Management," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S2

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6/18/2009	ML091560352	AP1000 Request For Withholding Information From Public Disclosure (DCP/NRC2463).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP/NRC2463
6/18/2009	ML091680608	Safety Evaluation Report With Open Items For Chapter 11, "Radio Active Waste Management," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S2
6/18/2009	ML091050516	Audit Report for the Review of Phase I and II Proprietary Documents related to the Software Lifecycle for the Protection and Safety Monitoring System for the AP1000.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/18/2009	ML091050554	Final Safety Evaluation Report for Section 3.6.3 Regarding the Westinghouse AP1000 Revision 17 Design Center Amendment.	Terao, D.	NRC/NRO/DE/CIB1	
6/19/2009	ML091740119	Letter - Safety Evaluation Report with Open Items for Chapter 12, "Radiation Protection," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL	NUREG-1793 S2
6/19/2009	ML091380100	Enclosure - Safety Evaluation Report with Open Items for Chapter 12 of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S2

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6/22/2009	ML091541130	Enclosure - Proprietary Safety Evaluation Report With Open Items For Chapter 10, "Steam And Power Conversion System," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO/DNRL/ NWE2	NUREG-1793 S2
6/22/2009	ML091550705	Safety Evaluation Report With Open Items For Chapter 10, "Steam And Power Conversion System," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S2
6/22/2009	ML091600421	Advisory Committee on Reactor Safeguards AP1000 Subcommittee Review of Selected Chapters of the Safety Evaluation Report with Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
6/23/2009	ML091751196	Letter - Safety Evaluation Report With Open Items for Chapter 1, "Introduction and General Discussion," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S2
6/23/2009	ML091751197	Enclosure - Safety Evaluation Report With Open Items for Chapter 1, "Introduction and General Discussion," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO/DNRL	NUREG-1793 S2

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6/23/2009	ML091751007	Appendix B. Pre-Application Chronology of Principal Actions, December 2004 - May 2007.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/23/2009	ML091760085	Appendix A. AP1000 Historical Perspective.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	AP1000
6/24/2009	ML091761017	Audit Plan to Review AP1000 DCD Revision 17 and for Chapter 9.	Segala, J. P.	NRC/NRO/DSRA/ SBPA	
6/24/2009	ML091520184	Audit Plan to Review AP1000 DCD Revision 16 and 17 and RTNSS/Cold Shutdown Issues for Chapter 9.	Segala, J. P.	NRC/NRO/DSRA/ SBPA	
6/24/2009	ML092120305	04/28/2009 Meeting Minutes, Attendance List and Slides, QA/ITAAC Workshop Conducted at the Westinghouse Energy Center in Monroeville, Pennsylvania.	Sisk, R. B.	Westinghouse	DCP_NRC_002512
6/25/2009	ML091800463	AP1000 Request For Withholding Information From Public Disclosure (CAR 080084, CAR 080093, PV-CEDM-CGI-001, NQCP-1501, NQCP-1602, NCR-080026, NCR-080448, NCR-080530, QCP-1801, RCA-080084 R1, RCA-080084 R3 and VER-0605-06).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	CAR 080084 CAR 080093 NCR-080026 NCR-080448 NCR-080530 NQCP-1501 NQCP-1602 PV-CEDM-CGI-001 QCP-1801 RCA-080084 R1 RCA-080084 R3 VER-0605-06
6/26/2009	ML091801089	Westinghouse Letter Regarding Draft Chapter 1 SER Review and no Proprietary Information Included in the SER.	Sisk, R. B.	Westinghouse	DCP_NRC_002580

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6/30/2009	ML091900166	07/14/2009 Public Meeting With Westinghouse On AP1000 Design Certification Amendment Structural Review Of Shield Building.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
7/1/2009	ML091900167	07/16/2009 Notice of Meeting with Westinghouse to Discuss AP1000 Design Control Document (DCD) Chapter 7, "Instrumentation And Control Systems", Open Issues and The Audit Report.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
7/2/2009	ML092030488	Submittal of Proprietary Presentation Materials from the June 15 and 16, 2009 Meeting Regarding AP1000 Shield Building Design.	Sisk, R. B.	Westinghouse	AP1000 AW-09-2610 DCP_NRC_002547
7/8/2009	ML091190773	2009/07/08 AP-1000 DCD Review - AP1000 DCD Capture - 7/8/09		NRC/NRO	
7/9/2009	ML091280332	Letter, Safety Evaluation Report With Open Items for Chapter 2, "Site Characteristics," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL	NUREG-1793, Suppl 2
7/9/2009	ML091280332	Letter, Safety Evaluation Report With Open Items for Chapter 2, "Site Characteristics," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL	NUREG-1793, Suppl 2

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7/9/2009	ML093170672	Enclosure - Proprietary Information Report Final Safety Evaluation Report with Open Items for Chapter 2, "Site Characteristics," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL	
7/9/2009	ML091980043	Enclosure - Proprietary Information Report Final Safety Evaluation Report with Open Items for Chapter 2, "Site Characteristics," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL	
7/10/2009	ML092230121	Transmittal of Proprietary and Non-Proprietary Reports Related to Turbine Reliability.	Sisk, R. B.	Westinghouse	AW-09-2611 DCP_NRC_002549 WCAP-16651-NP, Rev 1 WCAP-16651-P, Rev 1
7/13/2009	ML092230123	Draft Chapter 10 SER and Draft Chapter 11 SER Review.	Sisk, R. B.	Westinghouse	DCP_NRC_002559
7/16/2009	ML091960276	AP1000RCP-06-009-NP, Revision 2, "Structural Analysis Summary for the API 000 Reactor Coolant Pump High Inertia Flywheel."		Curtiss-Wright Flow Control Corp	
7/16/2009	ML091980326	AP1000RCP-06-009-P, Revision 2, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel."		Curtiss-Wright Flow Control Corp	
7/17/2009	ML092050141	AP1000 Request For Withholding Information From Public Disclosure (DCP/NRC2463).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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7/17/2009	ML092050142	2009/07/17 AP-1000 DCD Review - AP1000 DCD Capture - 7/17/09		NRC/NRO	
7/21/2009	ML092050158	Confirmatory Items AP1000 Design Certification Amendment Safety Evaluation Report Chapter 5.	Sisk, R. B.	Westinghouse	DCP_NRC_002561
7/21/2009	ML092030540	Open Items AP1000 Design Certification Amendment Safety Evaluation Report Chapter 5.	Sisk, R. B.	Westinghouse	DCP_NRC_002563
7/21/2009	ML092050652	Chapter 2 Safety Evaluation Report with Open Items Proprietary Review.	Sisk, R. B.	Westinghouse	DCP_NRC_002568
7/22/2009	ML091770646	2009/07/22 AP-1000 DCD Review - AP1000 DCD Capture - 7/22/09		NRC/NRO	
7/24/2009	ML092110534	Audit Report From June 25, 2009 to Review AP1000 Revision 16 and 17 and SFPCS Issues for Chapter 9.	Segala, J. P.	NRC/NRO/DSRA	
7/28/2009	ML092090295	NRC Response to AP1000 Request For Withholding Information From Public Disclosure (DCP/NRC2508).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
7/28/2009	ML092120340	Westinghouse Draft Chapter 12 SER Review.	Sisk, R. B.	Westinghouse	DCP_NRC_002566
7/30/2009	ML092190900	Safety Evaluation Report for AP1000 Design Certification Revision 17.	Thomas, B. E.	NRC/NRO/DE/SEB1	
7/31/2009	ML092190901	2009/07/31 AP-1000 DCD Review - AP1000 DCD Capture - 7/31/2009		NRC/NRO	

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7/31/2009	ML092190902	WCAP-17028-P, Rev. 1, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Appendix C.9, Page C-141 to End.	Andreychek, T. S. Byers, W. A. McKinley, J. K. McNamee, K. F. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002 DCP_NRC_002582
7/31/2009	ML092190903	Westinghouse, LLC - Transmittal of IRWST and CR Screen Related Documents.	Sisk, R. B.	Westinghouse	DCP_NRC_002582
7/31/2009	ML092190904	APP-GW-GLE-002, Rev. 3, "Impacts to the AP1000 to Address Generic Safety Issues (GSI)-191."		Westinghouse	DCP_NRC_002582
7/31/2009	ML092190905	APP-GW-GLR-079, Rev. 4, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."		Westinghouse	DCP_NRC_002582
7/31/2009	ML092190906	WCAP-17028-NP, Rev. 1, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents."	Andreychek, T. S. Byers, W. A. McKinley, J. K. McNamee, K. F. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002 DCP_NRC_002582
7/31/2009	ML092190921	APP-GW-GLR-079, Rev. 4, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."		Westinghouse	DCP_NRC_00258
7/31/2009	ML092150664	WCAP-17028-P, Rev. 1, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Cover Page Through Chapter 10.	Andreychek, T. S. Byers, W. A. McKinley, J. K. McNamee, K. F. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002 DCP_NRC_002582

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7/31/2009	ML092190396	WCAP-17028-P, Rev. 1, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Appendix A through C, Page C-140.	Andreychek, T. S. Byers, W. A. McKinley, J. K. McNamee, K. F. Schulz, T. L. Song, Y. J.	Westinghouse	DCP_NRC_002582
8/3/2009	ML092220043	Regulatory Audit Report - AP1000 Design Certification Amendment Section 3.9.3 on 10/13/08 - 10/17/08.	Le, T. D.	NRC/NRO/DE/EMB1	
8/4/2009	ML092180850	Westinghouse Response to Request for Additional Information on SRP 15 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC-002579
8/4/2009	ML092230120	Comment (8) of Robert Sisk, on Behalf of Westinghouse, on Draft Regulatory Guide DG-1191.	Sisk, R. B.	Westinghouse	74FR26440 00008 DCP_NRC_002585 DG-1191
8/6/2009	ML092230122	Audit Report from June 25, 2009 to Review AP1000 Revisions 16 and 17 and RTNSS/Cold Shutdown Issues for Chapter 9.	Segala, J. P.	NRC/NRO/DSRA/ SBPA	
8/7/2009	ML091800255	06/15/09 - 06/16/09 Summary of Public Meeting to Discuss Shield Building Design in the AP1000 Design Certification Amendment with Westinghouse.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
8/11/2009	ML092670395	Westinghouse - Open Item AP1000 Design Certification Amendment Safety Evaluation Report Chapter 11.	Sisk, R. B.	Westinghouse	DCP_NRC_002591

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8/12/2009	ML092100670	AP1000 Request for Withholding Information from Public Disclosure (DCP-NRC-002536).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP-NRC-002536
8/13/2009	ML092380361	08/25/2009 - GSI-191/Sump Meeting With Westinghouse On The AP1000 Design Certification Amendment.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	
8/18/2009	ML092380362	Results of Qualification Panel Examination for Wesley Held as a New Reactor Project Manager.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/18/2009	ML092360182	Westinghouse Submittal of APP-OCS-J1-002, Revision 1 "AP1000 Human System Interface Design Guidelines," Responses Related to RAI-SRP18-COLP-08	Sisk, R. B.	Westinghouse	DCP_NRC_002594
8/18/2009	ML092220136	APP-OCS-J1-002, Rev 1, "AP1000 Human System Interface Design Guidelines."		Westinghouse	DCP_NRC 2594
8/19/2009	ML092320092	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-137, Revision 0.	Sisk, R. B.	Westinghouse	APP-GW-GLR-137, Rev 0 DCP_NRC_002597
8/20/2009	ML092380077	Safety Evaluation Regarding AP1000 DCD Revision 17 Changes To Section 8.0, "Electric Power Systems".	Jenkins, R. V.	NRC/NRO/DE/EEB	
8/20/2009	ML092180345	09/02/2009-Notice of GSI-191/Sump Meeting With Westinghouse on the AP1000 Design Certification Amendment.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	GSI-191

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8/27/2009	ML092450703	Letter to R. Sisk, Westinghouse on NRC's Chapter 6 Review Progress and Schedule Impact for AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
8/31/2009	ML092450723	Audit Plan To Review AP1000 Design Control Document Revision 16 And 17 and Fuel Rack Seismic Issues for Chapter 9.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
8/31/2009	ML092450724	Westinghouse Response to Request for Additional Information on SRP Section 18 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002610
8/31/2009	ML092510094	Westinghouse Submittal of Revision 0 of APP-GW-E1-006 "AP1000 Cyber Security Design Criteria."	Sisk, R. B.	Westinghouse	DCP_NRC_002608
8/31/2009	ML092510095	Technical Document APP-GW-E1 -006, Revision 0, "AP 1000 Cyber Security Design Criteria".	Sisk, R. B.	Westinghouse	APP-GW-E1-006, Rev 0 DCP_NRC_2608
8/31/2009	ML092510096	Westinghouse, LLC - Submittal of "Design Methodology for AP1000 Enhanced Shield Building" - dated August 31, 2009 (APP-1200-S3R-003, Rev 0).	Sisk, R. B.	Westinghouse	AP1000 APP-1200-S3R-003, Rev 0 DCP_NRC_002606
9/2/2009	ML093270025	09/02/09 GSI-191/Sump Meeting Between NRC And Westinghouse on the AP1000 Design Certification Amendment.		Westinghouse	

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9/3/2009	ML092360296	AP1000 Request for Withholding Information from Public Disclosure for AP100RCP-06-009 Revision 2 (DCP-NRC-002579).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	AP1000RCP-06-009 Rev 2 DCP-NRC-002579
9/3/2009	ML092360781	09/03/09, Slides, Meeting Summary, DCWG Re: Implementation of DC/COL-ISG-08 "Necessary Content of Plant-Specific Technical Specifications."		NRC/NRO/DNRL/ NWE1	
9/4/2009	ML091980272	Propriety Information Review - Updated Safety Evaluation Report With Open Items For Chapter 8, "Electrical Power System," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
9/4/2009	ML092180710	Updated Safety Evaluation Report With Open Items For Chapter 8, "Electrical Power System," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO/DNRL/ NWE2	
9/8/2009	ML092180956	AP1000 Request For Withholding Information From Public Disclosure (APP-GW-GLR-065, Rev. 1).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	APP-GW-GLR-065, Rev 1
9/8/2009	ML092470207	Summary of a Category 1 Public Meeting Held with Westinghouse Regarding Proposed AP1000 Shield Building Design Methodology, Held In Rockville, Maryland on July 14, 2009.	Gleaves, B. C. Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	

Document Date	Accession Number	Title	Author Name	Author Affiliation	Case/Reference Number
9/9/2009	ML092510259	Letter, Safety Evaluation Report With Open Items For Chapter 18, "Human Factors Engineering," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
9/9/2009	ML092530359	Safety Evaluation Report With Open Items for Chapter 18, "Human Factors Engineering," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
9/9/2009	ML092940454	Federal Register Notice. Regarding the ACRS Subcommittee Meeting on AP 1000, October 6, 2009..		NRC/ACRS	
9/9/2009	ML092590314	On-Site Audit Review Plan for the AP1000 Sump Screen Downstream Effects/Long-Term Cooling Analyses.	Donoghue, J. E.	NRC/NRO/DSRA/ SRSB	AP1000
9/9/2009	ML092260313	APP-GW-J0H-001, Rev. 1, "AP1000 I&C Systems Cyber Security Plan," October 2009.	Batson, S. J.	Westinghouse	DCP_NRC_002663
9/14/2009	ML092300329	Notification of Proprietary Status, Submittal of "Design Methodology for AP1000 Enhanced Shield Building" - dated August 31, 2009 (APP-1200-S3R-003 Rev 0).	Sisk, R. B.	Westinghouse	APP-1200-S3R-003, Rev 0 DCP_NRC_002618

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9/15/2009	ML092540088	Letter - Safety Evaluation Report with Open Items for Chapter 3, Not Including Sections 3.7 Or 3.8, Titled "Design of Structures, Components, Equipment, and Systems," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
9/15/2009	ML092540264	Enclosure- Safety Evaluation Report With Open Items for Chapter 3, Not Including Sections 3.7 or 3.8, Titled "Design Of Structures, Components, Equipment, and Systems," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
9/15/2009	ML092580148	Letter, Safety Evaluation Report With Open Items For Chapter 13, Not Including Section 3.6, Titled "Conduct of Operations," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S02
9/15/2009	ML092470531	Safety Evaluation Report With Open Items for Chapter 13, Not Including Section 3.6, Titled "Conduct of Operations," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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9/15/2009	ML092510227	10/05/09 Notice of Meeting with Westinghouse On AP1000 Design Certification Amendment - Shield Building Structural Review.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	
9/17/2009	ML092510227	Phase 2b Safety Evaluation Report with Open Item for AP1000 Design Certification Document, Revision 17, and Section 15.3.	Lauron, C. L.	NRC/NRO/DSER/ RSAC	TAC RB5786
9/17/2009	ML092640647	Phase 2B SER Input with Open Item AP1000 Design Certification Amendment DCD Revision 17.		NRC/NRO/DSER/ RSAC	
9/17/2009	ML092650345	Phase 2B SER Input with Open Item AP1000 Design Certification Amendment DCD Revision 17.		NRC/NRO/DSER/ RSAC	
9/17/2009	ML092470453	Westinghouse Responses to Requests for Additional Information on Technical Report No. 54 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002624
9/17/2009	ML092540421	Westinghouse - AP1000 Response to Request for Additional Information Nos. RAI-SRP3.2.1-EMB2-06 & RAI-SRP3.8.2-CIB1-01 R2, in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002624 SRP 3

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9/18/2009	ML092610380	07/16/2009 Meeting Summary, Category 1 Public Meeting (Part of the Meeting Was) with Westinghouse Regarding AP1000 Design Control Document Chapter 7, "Instrumentation and Control System", Open Issues and the Audit Report.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
9/21/2009	ML092640230	Advisory Committee On Reactor Safeguards AP1000 Subcommittee Review of Selected Chapters of the Safety Evaluation Report with Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
9/21/2009	ML092640230	10/08/2009 Notice of Meeting with Industry's AP1000 Working Group to Discuss the Implementation of DC/COL-ISG-O8, "Necessary Content of Plant-Specific Technical Specification," Option 3.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
9/21/2009	ML092720624	10/05/09 Canceled Notice of Public Meeting with Westinghouse to Discuss the Westinghouse AP1000 Design Certification Amendment Building Structural .Review.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
9/21/2009	ML092610551	10/05/09 Canceled Notice of Public Meeting with Westinghouse to Discuss the Westinghouse AP1000 Design Certification Amendment Building Structural .Review.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	

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9/24/2009	ML093130436	AP1000 Design Certification Amendment: Safety Evaluation Report Chapter 10 - Status of Open Items.	Sisk, R. B.	Westinghouse	DCP_NRC_002636
9/25/2009	ML093170675	Summary of a Public Meeting with Westinghouse Regarding Generic Safety Issue 191 Held in Rockville, Maryland on March 16, 2009.	Donnelly, P. B.	NRC/NRO/DNRL/ NWE2	
9/29/2009	ML093170676	"Westinghouse Diverse Actuation System (DAS) Automatic Signal Setpoint Methodology."	Anderson, R. G. Ewald, J. G. Kindred, T. A. Monahan, J. S. Schulz, T. L. Stiffler, C. D. Tuley, C. R. Williams, M. G.	Westinghouse	
9/29/2009	ML093170677	Submittal of Proprietary and Non-Proprietary Reports Related to Spent Fuel Storage Racks Criticality Analysis APP-GW-GLR-029 R2.	Sisk, R. B.	Westinghouse	APP-GW-GLR-029, Rev 2 AW-09-2673 DCP_NRC_002623
9/30/2009	ML092950497	U.S. Nuclear Regulatory Commission AP1000 Spent Fuel Pool On-Site Audit May 5-7, 2009.	Donoghue, J. E.	NRC/NRO/DSRA/ SRSB	
9/30/2009	ML092950498	Letter Documenting Results of 03/11/2009 Telecom Relative to Chapter 14 Open Items from SER.	Sisk, R. B.	Westinghouse	DCP_NRC_002534
9/30/2009	ML092730229	U.S. Nuclear Regulatory Commission AP1000 Spent Fuel Pool On-Site Audit May 5 - 7, 2009.	Donoghue, J. E.	NRC/NRO/DSRA/ SRSB	
10/1/2009	ML092920045	Audit Report for AP1000 DCD Revision 17 Fuel Rack Seismic Issues for Chapter 9 - August 6 and 7, 2009.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	AP1000

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10/1/2009	ML092800157	AP1000 Request for Withholding Information from Public Disclosure (Table of Contents (TOC), Design Report for AP1000 Enhanced Shield Building (SB) APP-1200-S3R-003, Revision D and Shield Building Experimental Program, Keith Coogler, July 14, 2009).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	APP-1200-S3R-003, Rev D FOIA/PA-2010-0034
10/2/2009	ML092800158	Part 810 Westinghouse Suppliers (India) Incoming Letter to Interagency.	Goorevich, R. S.	US Dept of Energy, National Nuclear Security Admin	
10/5/2009	ML092670180	Westinghouse has Determined that Draft Chapter 8 SER Is Non-Proprietary in Nature and a Revision is Not Needed.	Sisk, R. B.	Westinghouse	DCP_NRC_002646
10/5/2009	ML092650266	Westinghouse Review of Chapter 3 Safety Evaluation Report with Open Items Proprietary Review.	Sisk, R. B.	Westinghouse	DCP_NRC_002645
10/7/2009	ML092670393	AP1000 Request for Withholding Information From Public Disclosure (APP-GEW-GLR-079, Rev. 4, WCAP-17028, Rev. 1).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	APP-GW-GLR-079, Rev 4 WCAP-17028, Rev 1
10/8/2009	ML092670397	Summary of Regulatory Audit Held with Westinghouse Regarding the Proposed AP1000 Design Certification Amendment, Cranberry Township, Pennsylvania, August 10-14, 2009.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	
10/12/2009	ML092860106	Enclosure 3 - Westinghouse Markup of NRC Chapter 18 SER Indicating Proprietary Sections.		Westinghouse	DCP_NRC_002649

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10/12/2009	ML092320205	Letter re: Chapter 18 SER Review.	Sisk, R. B.	Westinghouse	DCP_NRC_002649
10/13/2009	ML092880347	10/28/2009 Public Meeting With Westinghouse On AP1000 Design Certification Amendment - Shield Building Structural Review.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
10/15/2009	ML092880421	10/15/2009 Letter to Westinghouse From Dave Matthews to Rob Sisk regarding AP1000 Shield Building Design."	Matthews, D. B.	NRC/NRO/DNRL	
10/15/2009	ML092940445	Westinghouse Letter Enclosure - Details of Key Technical Issues Resulting from NRC Staff Review of AP1000 Shield Building Design Report APP-1200-SR3-003, Rev. 0, (Proprietary).	Matthews, D. B.	NRC/NRO/DNRL	
10/15/2009	ML092940453	Press Release-09-173: NRC Informs Westinghouse of Safety Issues with AP1000 Shield Building.		NRC/OPA	
10/15/2009	ML092800350	Westinghouse's Review of Chapter 13 Safety Evaluation Report.	Sisk, R. B.	Westinghouse	DCP_NRC_002662
10/16/2009	ML093090338	Submittal of "AP1000 I&C Systems Cyber Security Plan" - October 2009 (APP-GW-J0H-001 Rev 1).	Sisk, R. B.	Westinghouse	APP-GW-J0H-001, Rev 1 AW-09-2690 DCP_NRC_002663

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10/19/2009	ML093090339	Safety Evaluation Report With Open Items For Chapter 9, Not Including Sections 9.1.1.2.1 And 9.1.2.2.1 Fuel Rack Design Change Seismic Evaluations, Titled "Auxiliary Systems," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	10/19/2009
10/19/2009	ML093010261	Shaw Module Fabrication Schedule Presented to NRC at the Meeting in Westinghouse Cranberry PA Office on 09/10/2009.	Cronan, C.	Shaw Group, Inc.	
10/19/2009	ML092940320	Shaw Module Fabrication Schedule Presented to NRC at the Meeting in Westinghouse Cranberry PA Office on 09/10/2009 (Proprietary).		Shaw Group, Inc.	
10/20/2009	ML092990264	AP1000 Fuel Assembly Debris Testing - Remaining Tests, Revision A.	Schulz, T. L.	Westinghouse	AW-09-2692 DCP_NRC_002670
10/21/2009	ML093010260	2009/10/21 BEL COL - summary of 10/20/09 internal meeting to discuss language that was removed from Bellefonte SER regarding criteria to be used in future COL amendment reviews		NRC/NRO	
10/21/2009	ML093280183	Westinghouse Submittal of Non-Proprietary Version of the AP1000 Shield Building Design Report, APP-1200-SR3-003, Revision 0.	Sisk, R. B.	Westinghouse	DCP_NRC_002671

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10/21/2009	ML092800251	Transmittal of Proprietary Information relating to AP1000 Fuel Assembly Debris Testing for the Containment Debris Issue (GSI-191).	Sisk, R. B.	Westinghouse	AW-09-2692 DCP_NRC_002670
10/21/2009	ML092800266	AP1000 Design Certification Amendment Application - Response to Proposed Open Item (Chapter 3).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002695
10/23/2009	ML092960236	Letter - Safety Evaluation Report With Open Items For Chapter 7, "Instrumentation And Control," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
10/23/2009	ML093000519	Enclosure - Safety Evaluation Report With Open Items For Chapter 7, "Instrumentation And Control," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793
10/23/2009	ML093010365	10/28/2009 Notice of CANCELED Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Structural Review.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
10/23/2009	ML092920430	Westinghouse AP1000, Administrative Changes to Tier 1 of AP1000 Design Control Document.	Sisk, R. B.	Westinghouse	DCP_NRC_002672
10/23/2009	ML093060431	Westinghouse, ITAACs in Modules Presentation on September 10, 2009.	Sisk, R. B.	Westinghouse	DCP_NRC_002635

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10/26/2009	ML092990381	Advisory Committee On Reactor Safeguards AP1000 Subcommittee Review of Selected Chapters of the Safety Evaluation Report With Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
10/27/2009	ML093010461	Transmittal Schedule of GSI-191 Related Documents.	Sisk, R. B.	Westinghouse	DCP_NRC_002674 GSI-191
10/28/2009	ML093080103	11/18/09-Notice of Closed Meeting with Westinghouse to Discuss the AP1000 Integrated System Validation Plan.	Donnelly, P. B.	NRC/NRO/DNRL/ NWE2	
10/29/2009	ML093080337	11/18/09 Notice of Public Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Structural Review.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
10/29/2009	ML093080102	Transmittal of Proprietary Information Relating to AP1000 DAS Setpoint Methodology.	Sisk, R. B.	Westinghouse	dcp_NRC_002675
10/29/2009	ML093080338	Transmittal of AP1000 Shield Building Briefing for NNSA Presentation on 10/30/09.	Sisk, R. B.	Westinghouse	AW-09-2698 DCP_NRC_002679
10/30/2009	ML093080339	Westinghouse, Withdrawal of AP1000 Technical Report APP-GW-GLR-126, Revision 0 (TR 126).	Sisk, R. B.	Westinghouse	APP-GW-GLR-126, Rev 0 DCP_NRC_002680
10/30/2009	ML093080157	AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report.	Sisk, R. B.	Westinghouse	APP-GW-J1R-004 DCP_NRC_002683

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11/4/2009	ML093130384	11/04/09 Summary of Meeting with Industry's AP1000 Working Group to Discuss the Implementation of DC/COL-ISG-08, "Necessary Content of Plant-Specific Technical Specification," Option 3.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
11/4/2009	ML093130385	DRAFT Summary of Category 2 Public Meeting with Industry's AP1000 Working Group to Discuss the Implementation of DC/COL-ISG-08, "Necessary Content of Plant-Specific Technical Specification," Option 3.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
11/4/2009	ML093130386	AP1000 Shield Building Briefing Presentations for the NRC Meeting on 11/18/2009.	Sisk, R. B.	Westinghouse	AW-09-2696 DCP_NRC_002687
11/5/2009	ML093090193	Safety Evaluation Report Input for Tier 1 "Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC)" for Westinghouse Amendment, Docket No. 52-006.	Kowal, M. G.	NRC/NRO/DCIP/CTSB	
11/5/2009	ML093160601	Status Report for Revision 17 to AP1000 Design Control Document: Selected Chapters on November 5-7, 2009.	Lee, M. P.	NRC/ACRS	
11/5/2009	ML092730466	2009/11/05 AP-1000 DCD Review - AP1000 DCD Capture 11/5/09		NRC/NRO	
11/6/2009	ML093070733	LTR-09-0550 - Ltr. Edward Markey, re: AP1000 Nuclear Reactor Design...Safety Requirements Impact on Loan Guarantee Applications.	Markey, E. J.	US HR, Comm on Energy & Commerce	LTR-09-0550

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11/10/2009	ML093090260	Memo: Transmittal of TRACE AP1000 LBLOCA, SBLOCA, Transient, and Applicability Reports.	Uhle, J. L.	NRC/RES/DSA	
11/12/2009	ML093160715	Summary of the Audit Conducted at Westinghouse Regarding AP1000 Computer Based Procedures Held in Monroeville, PA On September 15-17, 2009.	Pieringer, P. A.	NRC/NRO/DCIP	
11/12/2009	ML093210485	10/21/09 Summary of Category 2 Public Meeting with the Nuclear Energy Institute to Discuss Health Physics Issues for New Reactors.	Kellner, R.	NRC/NRO/DCIP/CHPB	
11/12/2009	ML093210509	SER - Safety Evaluation Report With Open Items for Chapter 18, "Human Factors Engineering," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
11/13/2009	ML093340031	G20090651/LTR-09-0564/EDATS: SECY-2009-0512 - Ltr Mario V. Bonaca re: Westinghouse AP1000 Design Certification Amendment: Status of ACRS Review.	Bonaca, M. V.	NRC/ACRS	G20090651 LTR-09-0564 SECY-2009-0512
11/16/2009	ML093340032	Process For Closure Of Open Items And Confirmatory Items In Safety Evaluation Reports.	Akstulewicz, F. M.	NRC/NRO/DNRL	
11/17/2009	ML093340033	AP1000 Shield Building Presentations for the NRC Meeting on 11-18-09.	Sisk, R. B.	Westinghouse	AW-09-2703 DCP_NRC_002692

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11/18/2009	ML093280184	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-026, Rev. 2 (TR 44).	Sisk, R. B.	Westinghouse	APP-GW-GLR-026, Rev 2 DCP_NRC_002693 TR 44
11/18/2009	ML093280185	AP1000 Design Certification Amendment Application - Response to Request for Additional Information (SRP 6).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002696 SRP 6
11/18/2009	ML093160785	AP1000 Design Certification Amendment Application - Response to Request for Additional Information (SRP 9).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002694 SRP 9
11/18/2009	ML093170178	APP-GW-GLR-026, Rev. 2, "New Fuel Storage Rack Structural/Seismic Analysis, Technical Report Number 44."		Westinghouse	DCP_NRC_002693
11/19/2009	ML093280186	AP1000 Request for Withholding Information from Public Disclosure for Chapter 18 SER (DCP-NRC-002649).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP-NRC-002649
11/20/2009	ML093280187	AP1000 Request for Withholding Information From Public Disclosure For AP1000 I & C Systems Cyber Security Plan (APP-GW-J0H-001, Rev. 1).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	APP-GW-J0H-001, Rev 1
11/20/2009	ML093440190	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-033, Revision 3 (TR 54).	Sisk, R. B.	Westinghouse	DCP_NRC_002697
11/20/2009	ML093270166	APP-GW-GLR-033, Revision 3, "Spent Fuel Storage Rack Structure/Seismic Analysis, Technical Report Number 54."		Westinghouse	DCP_NRC_002697

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11/22/2009	ML093270296	NRC Audit of the AP1000 Probabilistic Risk Assessment in Support of Design Certification Amendment Review.	Mrowca, L. A.	NRC/NRO/DSRA/ SPLA	
11/23/2009	ML093370157	Canceled 11/18/2009 - Closed Meeting On The AP1000 Integrated System Validation Plan With Westinghouse.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	
11/24/2009	ML093370162	12/09/09 Notice of Meeting to Review AP1000 Human Factors Engineering Integrated System Validation Plan (APP-OCS-GEH-320) With Westinghouse Design Certification.	Donnelly, P. B.	NRC/NRO/DNRL/ NWE2	
11/25/2009	ML093380097	Transmittal of APP-GW-GLR-079 Revision 6, (Proprietary) "AP 1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."	Sisk, R. B.	Westinghouse	APP-GW-GLR-079, Rev 6 AW-09-2706 DCP_NRC_002699
11/25/2009	ML093380098	Westinghouse - Transmittal of IRWST and CR Screen Related Documents (Letter 1).	Sisk, R. B.	Westinghouse	AP1000 AW-09-2705 DCP_NRC_002700
12/1/2009	ML093380307	12/15/09-Notice of Meeting to Discuss NRC's Review of the Engineered Safety Features Chapter of the AP1000 Design Certification Amendment With Westinghouse.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	

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12/1/2009	ML093280435	Transmittal of APP-GW-GLR-086 Revision 1 (Non-Proprietary) "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA" (Alternate Document No. APP-GW-GLR-079-NP, Revision 6).	Sisk, R. B.	Westinghouse	AP1000 APP-GW-GLR-086, Rev 1 DCP_NRC_002701
12/1/2009	ML093430095	Westinghouse - Revision to AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-102.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002703 OI-SRP19.0-SPLA-07 OI-SRP19.0-SPLA-13
12/2/2009	ML100200634	Request For Withholding Information From Public Disclosure, Shaw Module Fabrication Schedule.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
12/2/2009	ML093410571	Transmittal of WCAP-16914 Revision 3 and WCAP-17028 Revision 3 Proprietary and Non-Proprietary Reports Related to Recirculation Screen Testing.	Sisk, R. B.	Westinghouse	AW-09-2710 DCP_NRC_002702 WCAP-16914, Rev 3 WCAP-17028, Rev 3
12/3/2009	ML093290121	Westinghouse LLC Columbia Site.		- No Known Affiliation	
12/4/2009	ML093130501	Westinghouse Response to Open Item on Chapter 16 in Support of AP1000 Design Certification Amendment Application	Sisk, R. B.	Westinghouse	DCP_NRC_002709
12/7/2009	ML093160184	AP1000 Request for Withholding Information from Public Disclosure, AP1000 Shield Building Briefing for National Nuclear Safety Administration (NNSA) Presentation on October 30, 2009.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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12/9/2009	ML093240354	Safety Evaluation Report With Open Items for Chapter 15, Titled "Transient and Accident Analyzes," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
12/9/2009	ML093510060	Transmittal of Safety Evaluation Report with Open Items For Chapter 15, Titled "Transient and Accident Analyzes," of NUREG 1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
12/10/2009	ML093420437	Letter - Response to Requests for Withholding of Proprietary Information In Accordance with 10 CFR Part 2, Section 2.390.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
12/13/2009	ML093450780	Enclosure 1 to DCP_NRC_002718, OI-SRP12.3-CHPB-03, Relative to the Physical Protection of an AP1000 Nuclear Power Plant.		Westinghouse	DCP_NRC_002718
12/14/2009	ML093560882	12/21-22/2009 Notice of Closed Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Testing, December 21-22, 2009 (with non-proprietary Enclosure 1 only).	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	

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12/14/2009	ML093560890	Proprietary Enclosure 2 to December 21-22, 2009, Notice of Closed Meeting with WEC on AP1000 DCA Shield Building Testing & Benchmarks.		NRC/NRO	
12/14/2009	ML093520054	Westinghouse. - Response to Request for Additional Information on SRP Section 6 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002716 SRP 6
12/17/2009	ML093640995	Submittal of Proprietary and Non-Proprietary Versions of WCAP-16592 (APP-PMS-GER-001), Revision 1.	Sisk, R. B.	Westinghouse	AW-09-2708 DCP_NRC_002720
12/18/2009	ML093640996	NEI ITAAC Threshold Examples from 12/17/09 Public Meeting.		NRC/NRO/DCIP/CTSB	
12/21/2009	ML093640997	AP1000 Shield Building - Benchmarking, Analysis, Testing and Design Overview.	Winters, J. W.	Westinghouse	AW-09-2711 DCP_NRC_002721
12/22/2009	ML093420963	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Verification Of Water Sources For Long-Term Recirculation Cooling Following A Loss-Of-Coolant Accident (APP-GW-GLR-079, Revision 6).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	APP-GW-GLR-079, Rev 6
12/22/2009	ML100050186	09/25/2009-Summary Of The Audit Conducted At Westinghouse Regarding AP1000 Passive Containment Cooling Water Held In Rockville, MD.	Hayes, M.	NRC/NRO/DSRA/ SPCV	

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12/24/2009	ML100050187	Advisory Committee on Reactor Safeguards AP1000 Subcommittee Review of Selected Chapters of the Safety Evaluation Report with Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
12/30/2009	ML100050188	Submittal of Proprietary and Non-Proprietary Versions of WCAP-17179, Revision 0.	Sisk, R. B.	Westinghouse	AW-09-2717 DCP_NRC_002725 WCAP-17179, Rev 0
12/30/2009	ML100050189	WCAP-17179 -NP, Revision 0, "AP1000TM Component Interface Module Technical Report." (Non-Proprietary).	Tweedle, T. W.	Westinghouse	APP-GW-GLR-144 AW-09-2717 DCP_NRC_002725
12/30/2009	ML100050190	WCAP-17179-P, Revision 0, "AP1000TM Component Interface Module Technical Report." (Proprietary).	Tweedle, T. W.	Westinghouse	APP-GW-GLR-143 AW-09-2717 DCP_NRC_002725
12/30/2009	ML100050277	AP1000 Shield Building - In-Plane Behavior of Concrete Filled Steel Elements.	Sisk, R. B.	Westinghouse	DCP_NRC_002728
12/30/2009	ML100050280	Submittal of Proprietary and Non-Proprietary Versions of WCAP- 17184, Revision 0.	Sisk, R. B.	Westinghouse	AW-09-2719 DCP_NRC_002727
12/30/2009	ML100050341	Enclosure 4 - WCAP-17184 -NP, Revision 0, "AP1000TM Diverse Actuation System Planning and Functional Design Summary Technical Report." (Non-Proprietary).	Stiffler, C. D.	Westinghouse	APP-GW-GLR-146 AW-09-2719 DCP_NRC_002727
12/30/2009	ML100050343	Submittal of Proprietary and Non-Proprietary Versions of WCAP-16674, Revision 2 and WCAP-16675, Revision 3.	Sisk, R. B.	Westinghouse	AP1000 AW-09-2718 DCP_NRC_002726

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1/5/2010	ML100080058	Westinghouse, AP1000 Shield Building - Benchmarking, Analysis, Testing and Design Overview (Results), in Support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002730
1/5/2010	ML100082096	Westinghouse, Re-submittal of Spent Fuel Storage Racks Criticality Analysis, APP-GW-GLR-029, Revision 2.	Sisk, R. B.	Westinghouse	APP-GW-GLR-029, Rev 2 DCP_NRC_002732
1/5/2010	ML093480081	APP-GW-GLR-029NP, Rev 2 (Reformatted), "AP1000 Spent Fuel Storage Racks Criticality Analysis."		Westinghouse	DCP_NRC_002732 HI-2094327, Rev 1
1/5/2010	ML100070056	APP-GW-GLR-029, Rev 2 (Reformatted), "AP1000 Spent Fuel Storage Racks Criticality Analysis."		Westinghouse	DCP_NRC_002732 HI-2094327, Rev 1
1/7/2010	ML093560519	AP1000 Request for Withholding Information from Public Disclosure for AP1000 Fuel Assembly [Debris] Tests Summary Results.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
1/7/2010	ML100190224	Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, February 2-3, 2010.		NRC/ACRS	
1/8/2010	ML100190225	AP1000 Request For Withholding Information From Public Disclosure For AP1000 Response to Request for Additional Information on SRP Section 9 (SRP9.1.2) (DCP_NRC_002691).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002691

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1/8/2010	ML100120686	AP1000 Spent Fuel Pool NRC Audit Report Dated 9/30/09 - Proprietary Exclusions.	Sisk, R. B.	Westinghouse	AW-10-2721 DCP_NRC_002731
1/11/2010	ML101100021	Westinghouse Response to NRC Open Item on Chapter 18 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002729
1/12/2010	ML093480328	AP1000 Request for Withholding Information from Public Disclosure for AP1000 Response to Request for Additional Information (SRP7.1) (DCP-NRC-002705).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002705
1/12/2010	ML100070187	2010/01/12 AP-1000 DCD Review - RE: AP1000 - New Draft RAI - RAI-SRP6.4-SPCV-11 R1		- No Known Affiliation	
1/13/2010	ML100130818	AP1000 Request for Withholding Information from Public Disclosure for AP1000 Response to Request For Additional Information (SRP7.1) (DCP-NRC-002676).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002676
1/13/2010	ML100190217	12/09/2009-Summary of Public Meeting with Westinghouse Regarding AP1000 Human Factors Engineering Integrated System Validation Plan (APP-OCS-GEH-320) In Rockville, Maryland.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	
1/13/2010	ML100190218	Rev. Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, February 2-3, 2010.	Wen, P. C.	NRC/ACRS	

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1/13/2010	ML093631156	Westinghouse - Chapter 9 Safety Evaluation Report with Open Items Proprietary Review.	Sisk, R. B.	Westinghouse	AW-10-2721 DCP_NRC_002734 NUREG-1793
1/13/2010	ML100250873	Westinghouse AP1000 - Chapter 9 SER Proprietary Markup. (Proprietary)		Westinghouse	AP1000 AW-10-2723 DCP_NRC_002734 NUREG-1793
1/15/2010	ML100250874	01/28/10 - 01/29/10-Notice of Public Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Structural Review.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
1/20/2010	ML100040552	Re-submittal of Proposed Changes for AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	AW-10-2728 DCP_NRC_002733 DCP_NRC_002744
1/21/2010	ML100270766	Request for Withholding Information from Public Disclosure, WCAP-16914, Revision 3 (App-My03-T2c-001) "Evaluation of Debris Loading Head Loss for AP1000 Recirculation Screens and In-Containment Refueling Water Storage Tank Screens".	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
1/21/2010	ML100270767	Request For Withholding Information From Public Disclosure, WCAP-17028, Revision 3 (APP-MY03-T2C-002) "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents."	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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1/21/2010	ML100270768	Westinghouse. - Response to Open Item on Chapter 12 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002746
1/22/2010	ML100271131	AP1000 Response to Proposed Open Item (Chapter 16).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002747
1/22/2010	ML100271132	Westinghouse - AP1000 Shield Building Structural Review Information for the NRC Meeting on 1/28/10 - 1/29/10.	Sisk, R. B.	Westinghouse	AP1000 AW-10-2729 DCP_NRC_002745
1/22/2010	ML093570243	AP1000 Shield Building Structural Review Information for the NRC Meeting on 1/28/10 - 1/29/10, Benchmarking For Out-of-Plane Test (LSDYNA).	Mei, S. Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002745
1/22/2010	ML100050730	AP1000 Shield Building Structural Review Information for the NRC Meeting on 1/28/10 - 1/29/10, Westwall Axial Tension About 10,000 kips.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002745
1/25/2010	ML100140460	November 18, 2009, Summary of Public Meeting Held with Westinghouse Regarding Proposed AP1000 Shield Building Design Methodology.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	
1/26/2010	ML100270140	Request For Withholding Information From Public Disclosure, WCAP-16943 "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" Topical Report.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	WCAP-16943

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1/26/2010	ML100320294	01/26/10 Summary of The Audit Conducted At Westinghouse Regarding AP1000 Documents APP-GW-GLR-138 and APP-GW-GLR-139 Held In Rockville, MD on November 13, 2009.	Hayes, M. W.	NRC/NRO/DSRA/ SPCV	APP-GW-GLR-138 APP-GW-GLR-139
1/27/2010	ML102460491	2010/01/27 AP-1000 DCD Review - FW: Red-Line Strikeout Revision to AP1000 SER Chapter 11		NRC/NRO	
1/27/2010	ML100350166	AP1000 Shield Building Structural Review - Final Public Information for 1/28/10 Meeting.	Sisk, R. B.	Westinghouse	DCP_NRC_002751
1/29/2010	ML101550648	01/29/2010 Meeting - Enclosure 3: Proprietary Meeting Slides.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
2/2/2010	ML100261632	Westinghouse, Review of Chapter 15 SER.	Sisk, R. B.	Westinghouse	DCP_NRC_002763
2/2/2010	ML100280262	Transcript of ACRS Subcommittee on AP1000 - Open Session, February 2 , 2010, Pages 1-297.		NRC/ACRS	NRC-044
2/3/2010	ML101550682	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Containment Recirculation And IRWST Screen Design, Technical Report 147 (APP-GW-GLN-147, Revision 3) (DCP_NRC_002700).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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2/3/2010	ML100430341	02/08/2010 Closed Meeting Notice With Westinghouse on AP1000 Design Certification Amendment - Shield Building Construction And Inspection Plans.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
2/3/2010	ML100430709	Transcript of ACRS Subcommittee on AP1000 - Open Session, February 3, 2010, Page 1-190.		NRC/ACRS	NRC-044
2/12/2010	ML100540179	02/23/10, Closed Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Construction Inspection Plans.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
2/12/2010	ML100540393	02/08/10 Canceled - Closed Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Construction and Inspection Plans.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
2/12/2010	ML100540393	AP1000 Shield Building Construction and Inspection Update Information for the NRC Meeting on 2/23/10.	Sisk, R. B.	Westinghouse	AP1000 AW-10-2738 DCP_NRC_002767
2/16/2010	ML100500672	December 15 Meeting Presentation Material on AP1000 GSI-191 Issue Resolution.	Sisk, R. B.	Westinghouse	AP1000 AW-10-2721 DCP_NRC_002778 GSI-191
2/16/2010	ML100570361	2010/02/16 AP-1000 DCD Review - OI-SRP 5.4.1-CIB1-01 follow up questions (AP1000 RCP flywheel analysis)		NRC/NRO	
2/17/2010	ML100570366	AP1000 Response to Request for Additional Information (SRP 3).	Sisk, R. B.	Westinghouse	DCP_NRC_002779

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2/17/2010	ML100570376	AP1000 Shield Building Structural Review Information for the NRC Meeting on 1/28/10 - 1/29/10 (Final).	Sisk, R. B.	Westinghouse	AW-10-2740 DCP_NRC_002768
2/17/2010	ML100890666	G20100106/EDATS: OEDO-2010-0153 - Edwin M. Hackett Memo re: ACRS Review of Amendments to Previously Certified Reactor Designs	Hackett, E. M.	NRC/ACRS	EDATS: OEDO-2010-0153 G20100106 OEDO-2010-0153
2/17/2010	ML100980141	2010/02/17 AP-1000 DCD Review - Acknowledgement of RAI-SRP6.1.2-CIB1-01		- No Known Affiliation	
2/17/2010	ML100280667	2010/02/17 AP-1000 DCD Review - Acknowledgement of RAI's - RAI-SRP6.2.2-SPCV-31 & RAI-SRP6.2.2-SPCV-26 R1		- No Known Affiliation	
2/18/2010	ML100491399	2010/02/18 AP-1000 DCD Review - RE: OI-SRP 9.1.5-SBPB-01 Response of 11/11/09		NRC/NRO	
2/22/2010	ML100500713	Summary of Audit Conducted at Westinghouse Regarding AP1000 Documents Addressing Generic Safety Issue GSI-191 Held in Rockville, Maryland on November 18, 2009.	Hayes, M.	NRC/NRO/DSRA/ SPCV	
2/22/2010	ML100541723	03/05/2010-Notice of Closed Meeting With Westinghouse On The AP1000 Sump Design And Associated Test Program That Address Generic Safety Issue (GSI) - 191.	Donnelly, P. B.	NRC/NRO/DNRL/ NWE2	GSI-191

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2/22/2010	ML100541965	Summary of Audit Conducted at Westinghouse Regarding AP1000 Documents Addressing Generic Safety Issue (GSI)-191 Held In Rockville, Maryland on November 18, 2009.		NRC/NRO/DSRA/ SBCV	
2/23/2010	ML101100067	2010/02/23 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP11.3-CHPB-05 and RAI-SRP11.5-CHPB-05		NRC/NRO	
2/23/2010	ML101100070	2010/02/23 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.2.2-SRSB-25 R1		NRC/NRO	
2/23/2010	ML100560171	2010/02/23 AP-1000 DCD Review - Acknowledgement of RAI-SRP6.2.2-SRSB-25 R1		- No Known Affiliation	
2/23/2010	ML100600408	2010/02/23 AP-1000 DCD Review - Acknowledgement of RAI-SRP11.3-CHPB-05 and RAI-SRP11.5-CHPB-05		- No Known Affiliation	
2/25/2010	ML100600409	2010/02/25 AP-1000 DCD Review - FW: OI-SRP19.0-SPLA-07, -13, & -14		NRC/NRO	
2/25/2010	ML100610309	December 15, 2009 Public Meeting Presentation on Passive Filtration & Main Control Room Habitability.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002795
2/25/2010	ML100640574	Westinghouse, Responses to Open Item on Chapter 3 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002797

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2/26/2010	ML100640575	12/21-22/2009 Summary of Category 1 Public Meeting Held With Westinghouse Regarding the Proposed AP1000 Shield Building Design Methodology.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
2/26/2010	ML100640576	Transmittal of IRWST and CR Screen Related Documents.	Sisk, R. B.	Westinghouse	DCP_NRC_002796
2/28/2010	ML100700452	WCAP-17028-P, Revision 4, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Cover through Page 8-93.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002 DCP_NRC_002798
2/28/2010	ML100820177	WCAP-17028-P, Revision 4, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Page 8-94 through Page C-330.		Westinghouse	APP-MY03-T2C-002 DCP_NRC_002798
2/28/2010	ML100820178	WCAP-17028-P, Revision 4, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents," Page C-331 through Page E-7 (end).		Westinghouse	APP-MY03-T2C-002 DCP_NRC_002798
2/28/2010	ML100550420	WCAP-17201-NP, Rev. 0, "AC 160 Hugh Speed Link Communication Compliance to DI&C-ISG-04 Staff Positions 9, 12, 13 and 15 Technical Report."	McLaughlin, T. J. Odess-Gillett, W. R.	Westinghouse	DCP_NRC 002826

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2/28/2010	ML100700451	WCAP-17201-P, Rev. 0, "AC160 Hugh Speed Link Communication Compliance to DI&C-ISG-04 Staff Positions 9, 12, 13 and 15 Technical Report."	McLaughlin, T. J. Odess-Gillett, W. R.	Westinghouse	APP-GW-GLR-148 DCP_NRC 002826
3/1/2010	ML100630209	01/28/10 - 01/29/10 Summary of Meeting with Westinghouse Regarding the Proposed AP1000 Shield Building Design Methodology, Held In Rockville, Maryland.	Proctor, C. M.	NRC/NRO/DNRL/ NWE2	
3/1/2010	ML100630210	Transmittal of WCAP-17028-P, R4, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents."	Sisk, R. B.	Westinghouse	AW-10-2760 DCP_NRC_002798 WCAP-17028-P, Rev 4
3/2/2010	ML100630211	Westinghouse Response to Open Item on Chapter 7 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002800
3/2/2010	ML100640414	Westinghouse Response to Open Item on Chapter 8 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002801
3/2/2010	ML100640415	Westinghouse Response to Request for Additional Information on Technical Report No. 03 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002802
3/2/2010	ML101100063	AP1000 Shield Building Construction and Inspection Information for the NRC Meeting on 2/23/10 (FINAL).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002803

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3/2/2010	ML100640704	2010/03/02 AP-1000 DCD Review - Acknowledgement of RAI-SRP 3.9.4-EMB1-01		- No Known Affiliation	
3/3/2010	ML100601038	2010/03/03 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP 3.9.4-EMB1-01		NRC/NRO	
3/3/2010	ML100640740	Westinghouse Response to Open Item on Chapter 3 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002804
3/5/2010	ML100670562	03/17/10 - 03/18/10 Notice of Meeting with Westinghouse to Discuss AP1000 Design Certification Amendment - Design Changes to be Incorporated in Revision 18 to the Design Control.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
3/5/2010	ML100700204	2010/03/05 AP-1000 DCD Review - FW: Emailing: OI-SRP4.5.1-CIB1-01 pkg 05-13-09.pdf		NRC/NRO	
3/5/2010	ML100700207	03/17/10 - 03/18/10 Time Change Notice of Meeting with Westinghouse to Discuss AP1000 Design Certification Amendment - Design Changes to be Incorporated In Revision 18 to the Design Control Document.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
3/5/2010	ML100700460	Transmittal of AP1000 Containment Debris Fuel Assembly Testing and RAI Response Summary Presentation.	Sisk, R. B.	Westinghouse	DCP_NRC_002812

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3/5/2010	ML100700460	AP1000 Containment Debris Fuel Assembly Testing and RAI Response Summary.		Westinghouse	DCP_NRC_002812
3/5/2010	ML100740575	03/17-18/2010 Revised Notice of Meeting With Westinghouse On AP1000 Design Certification Amendment - Design Changes To Be Incorporated In Revision 18 To The Design Control Document.	Jaffee, D. H.	NRC/NRO/DNRL/ NWE2	
3/5/2010	ML100610459	03/17-18/2010 Revised Notice of Meeting With Westinghouse On AP1000 Design Certification Amendment - Design Changes To Be Incorporated In Revision 18 To The Design Control Document.	Jaffee, D. H.	NRC/NRO/DNRL/ NWE2	
3/5/2010	ML100740529	03/17/2010 - 03/18/2010 Revised Notice of Meeting With Westinghouse on AP1000 Design Certification Amendment - Design Changes to be Incorporated in Revision 18 to the Design Control Document.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
3/8/2010	ML101100055	02/23/2010 Summary of Closed Meeting With Westinghouse Electric Regarding the Proposed AP1000 Shield Building Construction Inspection Plan Held in Rockville, Maryland.	Proctor, C. M.	NRC/NRO/DNRL/ NWE2	
3/8/2010	ML100710275	2010/03/08 AP-1000 DCD Review - RE: RAI-SRP 5.2.3-CIB1-01 Revision 2		- No Known Affiliation	
3/8/2010	ML100690264	2010/03/08 AP-1000 DCD Review - Acknowledgement of RAI-SRP2.5-RGS1-21		- No Known Affiliation	

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3/9/2010	ML100920517	Transmittal of AP1000 Containment Debris Fuel Assembly Testing and RAI Response Summary Presentation.	Sisk, R. B.	Westinghouse	DCP_NRC_002816
3/10/2010	ML100980070	On-Site Audit Review Plan for the AP1000 Sump Screen Downstream Effects/Long-Term Cooling Analyses.	Donoghue, J. E.	NRC/NRO/DSRA/SRSB	
3/11/2010	ML100750037	2010/03/11 AP-1000 DCD Review - Acknowledgement of RAI-SRP2.2-RSAC-01		- No Known Affiliation	
3/11/2010	ML100750692	2010/03/11 AP-1000 DCD Review - RE: RESEND - OI-SRP 9.1.4-SBPB-03 Response of 10/15/09		NRC/NRO	
3/12/2010	ML100750693	Technical Report Review, AP1000 Response to Open Item (OI-SRP5.4.1-CIB1-01).		Westinghouse	DCP_NRC_002817 OI-SRP5.4.1-CIB1-01
3/12/2010	ML100750694	Supplementary Information to DCP_NRC_002744 - Re-submittal of Proposed Changes for AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002818
3/12/2010	ML100610207	Supplementary - Proposed Changes for AP1000 DCD Rev. 18.		Westinghouse	DCP_NRC_002818
3/12/2010	ML100610437	Design Control Document Page Mark Ups for Change Numbers 53 and 57.		Westinghouse	DCP_NRC_002818
3/15/2010	ML100610502	12/15/09 Summary of Meeting With Westinghouse Regarding The Review of Chapter 6 of The Design Certification Amendment Held In Rockville, Maryland on December 15, 2009.	Donnelly, P. M.	NRC/NRO/DNRL/NWE2	

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3/15/2010	ML100740531	Slides - Passive Filtration & Main Control Room Habitability.	Donnelly, P. B.	NRC/NRO/DNRL/ NWE2	
3/15/2010	ML100780110	Slides - AP1000 GSI-191 Issue Resolution Recent Informal NRC Questions.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	GSI-191
3/15/2010	ML100770110	2010/03/15 AP-1000 DCD Review - AP1000 DCD Capture - 3/15/10		NRC/NRO	
3/16/2010	ML100770212	Preliminary Agenda and Representative Draft Slides for Westinghouse Fuel Performance Update Meeting on April 13-14, 2010.		Westinghouse	LTR-NRC-10-16-P
3/17/2010	ML100770213	Audit Report from December 8, 2009 and January 28, 2010, to Review AP1000 Revision 16 and 17 and SFPCS Issues for Chapter 9.	Segala, J. P.	NRC/NRO/DSRA/ SBPA	
3/18/2010	ML100770241	2010/03/18 AP-1000 DCD Review - FW: RAI-SRP-5.2.3-CIB1-03 Rev. 1 - Supplemental RAI		- No Known Affiliation	
3/18/2010	ML100820174	2010/03/18 AP-1000 DCD Review - AP1000 DCD Capture - 3/18/2010		NRC/NRO	
3/18/2010	ML100820173	2010/03/18 AP-1000 DCD Review - AP1000 DCD Capture - 3/18/2010		NRC/NRO	
3/18/2010	ML100820176	10 CFR 50.46 Annual Report for the AP1000 Standard Plant Design.	Sisk, R. B.	Westinghouse	DCP_NRC_002824
3/19/2010	ML100820239	Closure Basis for Proprietary Status of AP1000 Shield Building Reports per 10 CFR 2.390.	Sisk, R. B.	Westinghouse	DCP_NRC_002825

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3/19/2010	ML100820240	Submittal of Proprietary and Non-Proprietary Versions of WCAP-17201, Revision 0.	Sisk, R. B.	Westinghouse	AW-10-2778 DCP_NRC_002826 WCAP-17201, Rev 0
3/19/2010	ML100820241	Enclosure 4 to DCP_NRC_002819 - Westinghouse AP1000 - DCD Revision 17 Changes (Post TR-134 Rev. 5) Matrix.		Westinghouse	DCP_NRC_002819
3/19/2010	ML100830504	Revision of Submittal of Change Matrix for APP-GW-GL-700 Revision 17, "AP1000 Design Certification Document."	Sisk, R. B.	Westinghouse	APP-GW-GL-700, Rev 17 AW-10-2778 DCP_NRC_002819
3/19/2010	ML100900294	Enclosure 3 to DCP_NRC_002819 - Westinghouse AP1000 - DCD Revision 17 Changes (Post TR-134 Rev.5) Matrix.		Westinghouserp	DCP_NRC_002819
3/19/2010	ML101200064	Westinghouse Response to Open Item on Chapter 19 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002828
3/19/2010	ML100630779	Transmittal of WCAP-17028, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents."	Sisk, R. B.	Westinghouse	DCP_NRC_002827
3/19/2010	ML100820226	AP1000 Safeguards Assessment Submittal (APP-GW-GLR-066, Revision 3).	Winters, J. W.	Westinghouse	APP-GW-GLR-066, Rev 3

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3/22/2010	ML100840079	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Response To Open Item OI-SRP7.2-ICE-01) Proprietary And Non-Proprietary (DCP_NRC_002792).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002792
3/22/2010	ML100840080	Urgency Request for RFPA; AP1000 - Review Sections 6.3 and 15.6.5.	Madden, P. M.	NRC/NRO/DNRL	
3/22/2010	ML100840081	Submittal of AP1000 Flow Skirt Testing 1/4 Scale Test Facility Presentation Material dated February 2 and 3, 2010.	Sisk, R. B.	Westinghouse	AW-10-2786 DCP_NRC_002831
3/22/2010	ML100850633	Shearon Harris, Units 2 & 3, Levy, Units 1 & 2 - Submittal of Annual Report for AP1000 Standard Plant Design.	Kitchen, R.	Progress Energy Carolinas, Inc	AP1000 NPD-NRC-2010-024
3/22/2010	ML100850634	APP-1200-S3R-003, Revision 1, "Design Report for the AP1000 Enhanced Shield Building."		Westinghouse	DCP_NRC_002835
3/22/2010	ML100850636	Submittal of "Design Report for the AP1000 Enhanced Shield Building" - dated March 22, 2010 (APP-1200-S3R-003 Rev. 1).	Sisk, R. B.	Westinghouse	AW-10-2781 DCP_NRC_002830
3/22/2010	ML100640499	APP-1200-S3R-003, Rev. 1, "Design Report for AP1000 Enhanced Shield Building," Cover through Page 7-45.		Westinghouse	DCP_NRC_002830
3/22/2010	ML100670312	APP-1200-S3R-003, Rev. 1, "Design Report for AP1000 Enhanced Shield Building," Page 7-46 through End.		Westinghouse	DCP_NRC_002830

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3/23/2010	ML100680151	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Shield Building Construction And Inspection Update Information For The NRC Meeting On February 23, 2010, (DCP_NRC_002767).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002767
3/23/2010	ML100700320	AP1000, Request For Withholding Information From Public Disclosure, AP1000 Shield Building Structural Review Information For The NRC Meeting On January 28, 2010, - January 29, 2010, (Final) (DCP_NRC_002768).	Clark, P. M. Jabbour, K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002768
3/23/2010	ML100750171	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Shield Building - Benchmarking, Analysis, Testing And Design Overview (Results) (DCP_NRC_002730).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002730
3/23/2010	ML100750190	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Shield Building - Benchmarking, Analysis, Testing and Design Overview (DCP_NRC_002721).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002721
3/24/2010	ML100770461	Summary of the Audit Conducted at Westinghouse Regarding AP1000 Revision 17 Section 3.5.1.4 Automobile Missile Generation Issue Held In Rockville, Maryland on February 24, 2010.	Segala, J. P. Thomas, B. E.	NRC/NRO/DE/SEB1 NRC/NRO/DSRA	

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3/24/2010	ML100850192	04/08/2010 Meeting Notice With Westinghouse to Discuss of Component Interface Module and Diverse Actuation System Technical Reports of the AP1000 Instrumentation and Control System.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
3/24/2010	ML100910305	Submittal of "Design Report for the AP1000 Enhanced Shield Building" - dated March 22, 2010 (APP-1200-S3R-003 Rev. 1).	Sisk, R. B.	Westinghouse	APP-1200-S3R-003, Rev 1 DCP_NRC_002835
3/26/2010	ML100680304	2010/03/26 AP-1000 DCD Review - FW: Acknowledgement of RAI SRP6.1.1-CIB1-01 R1		NRC/NRO	
3/29/2010	ML100690367	AP1000, Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Technical Document Information, Response to Request for Additional Information on Standard Review Plan Section 6.2.2 (DCP_NRC_002759).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002759
3/29/2010	ML100710226	AP1000 Request for Withholding Information From Public Disclosure, December 15 Meeting Presentation Material On AP1000 GSI-191 Issue Resolution (DCP_NRC_002778).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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3/29/2010	ML100890054	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Response To Open Item (OI-SRP7.9-Ice-01) Proprietary And Non-Proprietary (DCP_NRC_002756).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	AP1000 DCP_NRC_002756 OI-SRP7.9-ICE-01
3/29/2010	ML100890069	AP1000, Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Technical Document Information, Re-Submittal of Proposed Changes for AP1000 Design Control Document Revision 18 (DCP_NRC_002744).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002744
3/30/2010	ML100890071	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.2.2-CIB1-31	Butler, R.	NRC/NRO/DNRL	
3/30/2010	ML100890072	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.2.2-CIB1-30		NRC/NRO	
3/30/2010	ML100890186	2010/03/30 AP-1000 DCD Review - FW: Emailing: RAI-SRP17.4-SPLA-05 R1 pkg 03-23-10.pdf		NRC/NRO	
3/30/2010	ML100890649	2010/03/30 AP-1000 DCD Review - FW: Emailing: RAI COL03.05.01.04-1 R1 RAI-SRP3.7.1-SEB1-06 R4 OI-SRP3.3.2-SEB1-01 R1 OI-SRP3.7.2-SEB1-02 R1 pkg 03-24-10.pdf		NRC/NRO	

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3/30/2010	ML100890650	Completing Rulemaking on the AP1000 Amendment (Rev. 17).		Westinghouse	
3/30/2010	ML100890651	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.2-RSAC-01		NRC/NRO	
3/30/2010	ML100890652	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.5-RGS1-21		NRC/NRO	
3/30/2010	ML100890657	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.2.2-SRSB-25 R1		NRC/NRO	
3/30/2010	ML100890658	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.2.2-CIB1-28 R1		NRC/NRO	
3/30/2010	ML100890662	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.5-RGS1-21		NRC/NRO	
3/30/2010	ML100890665	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.2-RSAC-01		NRC/NRO	
3/30/2010	ML100890667	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of new RAI-SRP 5.4.1-CQVB-01 - variable speed drives		NRC/NRO	
3/30/2010	ML100900292	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.1.2-CIB1-01		NRC/NRO	

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3/30/2010	ML100920282	2010/03/30 AP-1000 DCD Review - FW: Acknowledgement of RAI's - RAI-SRP6.2.2-SPCV-31 & RAI-SRP6.2.2-SPCV-26 R1		NRC/NRO	
3/31/2010	ML100920285	WCAP-17028-NP, Rev. 05, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents."	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-004
3/31/2010	ML100920287	WCAP-17028-P, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Section 8.36 to Appendix C-88.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002
3/31/2010	ML100920289	WCAP-17028-P, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Appendix C-89 to C-270.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002
3/31/2010	ML100920291	WCAP-17028-P, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Cover to Section 8-99.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002

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3/31/2010	ML100920292	WCAP-17028-P, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Appendix C-271 to C-428.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002
3/31/2010	ML100950116	WCAP-17028-P, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Appendix C-429 to C-594.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002
3/31/2010	ML100970364	WCAP-17028-P, Rev. 5, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies During Loss of Coolant Accidents." Appendix C-595 to End.	Baier, S. L. Byers, W. A. Pezze, J. P. Ruth, K. L. Scaddozzo, G. Schulz, T. L. Song, Y. J.	Westinghouse	APP-MY03-T2C-002
4/1/2010	ML100920509	AP1000 Response to Proposed Open Item (Chapter 3), OI-SRP3.12-EMB-4 R1 and OI-SRP3.6-2-EMB2-01 R1, in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002845
4/2/2010	ML100920512	2010/04/02 AP-1000 DCD Review - Emailing: OI-SRP3.6.2-EMB2-01 R1 OI-SRP3.12-EMB-4 R1 pkg 04-01-10.pdf		NRC/NRO	
4/2/2010	ML100920513	2010/04/02 AP-1000 DCD Review - FW: Emailing: OI-SRP3.6.2-EMB2-01 R1 OI-SRP3.12-EMB-4 R1 pkg 04-01-10.pdf		NRC/NRO	

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4/2/2010	ML100920514	2010/04/02 AP-1000 DCD Review - FW: Transmittal of RAI SRP17.4-SPLA-04		NRC/NRO	
4/2/2010	ML100920515	2010/04/02 AP-1000 DCD Review - FW: Transmittal of RAI SRP17.4-SPLA-04		NRC/NRO	
4/2/2010	ML100920518	2010/04/02 AP-1000 DCD Review - FW: Emailing: RAI-SRP17.4-SPLA-05 R1		NRC/NRO	
4/2/2010	ML100950238	2010/04/02 AP-1000 DCD Review - FW: Emailing: RAI-SRP17.4-SPLA-05 R1		NRC/NRO	
4/2/2010	ML100950315	2010/04/02 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.2-RSAC-01		NRC/NRO	
4/5/2010	ML100950469	04/15/2010 Notice of Forthcoming Meeting with Westinghouse to Discuss Request for Additional Information SRP2.2-RSAC-01, Hydrogen Explosion and Other Chemical Hazard of the AP1000 Reactor Site.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
4/5/2010	ML101100023	Meeting Minutes of the AP1000 Subcommittee, October 6-7th, 2009 (Open).	Mitchell-Funderbun k, N.	NRC/ACRS	
4/5/2010	ML100570217	Meeting Minutes of the AP1000 July 23-24, 2009 Subcommittee Meeting (Open).	Wang, W.	NRC/ACRS	
4/5/2010	ML100980048	2010/04/05 AP-1000 DCD Review - RE: AP1000 - New draft RAI - RAI-SRP6.2.2-SPCV-32		- No Known Affiliation	
4/7/2010	ML100980072	Memo - Procedure For Chapter Validation And Integration.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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4/8/2010	ML100980143	Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, April 22, 2010.	Dias, A. F.	NRC/ACRS	
4/8/2010	ML101520248	2010/04/08 AP-1000 DCD Review - AP1000 DCD Capture - 4/8/10		NRC/NRO	
4/8/2010	ML101520340	2010/04/08 AP-1000 DCD Review - AP1000 DCD Capture - 4/8/10 2		NRC/NRO	
4/8/2010	ML100900234	AP1000 Diverse Actuation System (DAS) Setpoint Methodology – Westinghouse Responses to NRC Questions.		Westinghouse	
4/8/2010	ML100960417	Slides, 04/08/2010, Discussion on AP1000 Component Interface Module and Defense-in-Depth and Diversity Design.	Jackson, T. W.	NRC/NRO/DE/ICE1	
4/9/2010	ML100990094	04/09/2010 Summary of Category 1 Public Meeting With Westinghouse Regarding the Proposed AP1000 Design Change Packages.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
4/9/2010	ML101040347	New Reactor Information Slides.	Junge, M. A.	NRC/NRO/DCIP/CHPB	
4/9/2010	ML101090173	04/22/2010 Meeting Notice with Westinghouse to Discuss the Effects of High Frequency Seismic Ground Motion on the Seismic Margin of Ap1000 Structures, Systems, and Components.	Sanders, S.	NRC/NRO/DNRL/ NWE2	

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4/9/2010	ML101100022	Westinghouse Submittal of Revision 4 of AP1000 Standard Combined License Technical Report No. 3, "Extension of Nuclear Island Seismic Analysis to Soil Sites."	Sisk, R. B.	Westinghouse	DCP_NRC_002848
4/19/2010	ML101100024	04/26/2010 Revised Meeting Notice with Westinghouse to Discuss the Effects of High Frequency Seismic Ground Motion on the Seismic Margin of AP1000 Structures, Systems, and Components.	Sanders, S.	NRC/NRO/DNRL/ NWE2	
4/20/2010	ML101100026	2010/04/20 AP-1000 DCD Review - FW: AP1000 - New Draft RAI - RAI-SRP6.4-SPCV-11 R1		NRC/NRO	
4/20/2010	ML101100028	2010/04/20 AP-1000 DCD Review - FW: AP1000 - New draft RAI - RAI-SRP6.2.2-SPCV-32		NRC/NRO	
4/20/2010	ML101100054	2010/04/20 AP-1000 DCD Review - FW: AP1000 - New Draft RAIs - RAI-SRP6.4-SPCV-09-14		NRC/NRO	
4/20/2010	ML101100059	2010/04/20 AP-1000 DCD Review - FW: AP1000 - New Draft RAIs - RAI-SRP6.2.2-CIB1-24		NRC/NRO	
4/20/2010	ML101100066	2010/04/20 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.2-RSAC-01		NRC/NRO	
4/20/2010	ML101100068	2010/04/20 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP2.5-RGS1-21		NRC/NRO	

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4/20/2010	ML101100071	2010/04/20 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP 3.9.4-EMB1-01		NRC/NRO	
4/20/2010	ML101121005	2010/04/20 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP6.2.2-SRSB-25 R1		NRC/NRO	
4/20/2010	ML101121013	2010/04/20 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP11.3-CHPB-05 and RAI-SRP11.5-CHPB-05		NRC/NRO	
4/20/2010	ML101230513	Westinghouse Response to Request for Additional Information on SRP Section 17 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002853
4/21/2010	ML101100713	AP1000 Standard COL Technical Report Submittal of APP-GW-S2R-010, Revision 4 (TR3).	Sisk, R. B.	Westinghouse	DCP_NRC_002855

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4/21/2010	ML101160477	Petition to Initiate Special Investigation on Significant AP1000 Design Defect.	Runkle, J. D.	Bellefonte Efficiency & Sustainability Team (BEST) Blue Ridge Environmental Defense League Georgia Women's Action for New Directions Green Party of Florida North Carolina Waste Awareness & Reduction Network (NC WARN) Nuclear Information & Resource Service (NIRS) Nuclear Watch South Sierra Club Southern Alliance for Clean Energy	
4/23/2010	ML101160478	04/26/2010-Notice of Closed Meeting With Westinghouse On The AP1000 One Percent Power Uncertainty Error Issue.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
4/23/2010	ML101160479	Submittal of AP1000 Calorimetric Power Uncertainty April 26, 2010 Presentation.	Sisk, R. B.	Westinghouse Westinghouse	AW-10-2799 DCP_NRC_002857
4/26/2010	ML101050285	Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002850
4/26/2010	ML101180421	Description of Proposed Changes for AP1O0O DCD Rev. 18.		Westinghouse	DCP_NRC_002850

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4/27/2010	ML101060298	Audit Plan of AP1000 Design Certification - Regulatory Audit of Open Items: J Groove Welds and Containment Recirculation Screen.	Dixon-Herrity, J. L.	NRC/NRO/DE/EMB2	
4/28/2010	ML101190163	Inspection of Westinghouse Software Verification and Validation of the Macro Code for Design of Reinforced Concrete Structures.	Peralta, J. D.	NRC/NRO/DCIP/CQVP	
4/29/2010	ML101190165	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of AP1000 Shield Building Construction And Inspection Information For The NRC Meeting 2/23/10 (FINAL) (DCP_NRC_002803).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
4/29/2010	ML100960243	2010/04/29 AP-1000 DCD Review - RE: OI-SRP 9.1.3-SBPA-13 Rev. 1		- No Known Affiliation	
4/29/2010	ML100970293	2010/04/29 AP-1000 DCD Review - AP1000 DCD Capture - 4/29/2010		NRC/NRO	
4/30/2010	ML102000155	Advanced Final Safety Evaluation Report For Chapter 14 Titled "Verification Programs," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793
4/30/2010	ML102000156	Enclosure - Chapter 14. Verification Programs.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	NUREG-1793

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5/3/2010	ML101200263	04/15/2010 Summary of Category 1 Public Meeting With Westinghouse To Discuss Request For Additional Information SRP2.2-RSAC-01, Hydrogen Explosion and Other Chemical Hazard of The AP1000 Reactor Site.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
5/3/2010	ML101050410	Evaluation of Hydrogen Explosion and Other Chemical Hazards to the AP1000 Reactor Plant Westinghouse April 15, 2010.		NRC/NRO/DNRL/ NWE2	
5/3/2010	ML101270258	Inspection Plan for Westinghouse AP1000 Design Certification - Software Verification and Validation Inspection of Macro Stuctural Design Code.	Kavanagh, K. A.	NRC/NRO/DCIP/CQVP	
5/5/2010	ML101320112	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of WCAP-17201(APP-GW-GLR-148), Revision 0 (DCP_NRC_002826).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
5/7/2010	ML101450349	05/20/10-Meeting With Westinghouse on AP1000 Design Certification Amendment - Design Changes To Be Incorporated In Revision 18 To The Design Control Document.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	

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5/7/2010	ML101450350	Westinghouse, AP1000 Safeguards Assessment Resubmittal of Figures (APP-GW-GLR-066 Revision 3).	Winters, J.	Westinghouse	NPP_NRC_000002
5/7/2010	ML101450478	Submittal of "Design Report for the AP1000 Enhanced Shield Building" - dated May 7, 2010 (APP-1200-S3R-003 Rev 2).	Sisk, R. B.	Westinghouse	APP-1200-S3R-003, Rev 2 AW-10-2812 DCP_NRC_002865
5/7/2010	ML101450480	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building."		Westinghouse	DCP_NRC_002865
5/7/2010	ML101450481	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," Cover Page through Chapter 6, Section 6.3.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101450483	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," Chapter 7, Section 7.1 through Figure 7.11-23.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101450484	APP-1200-S3R-003, Revision 2, Design Report for the AP1000 Enhanced Shield Building," Chapter 7, Figure 7.11-24 through Chapter 8, Section 8.11.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101450485	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," Appendix B, Figure B-78 through Figure B-192.		Westinghouse	DCP_NRC_002865

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5/7/2010	ML101450486	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," Appendix C, Section C.1 through Figure C.4-26.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101450486	APP-1200-S3R-003, Revision 2, Design Report for the AP1000 Enhanced Shield Building," Chapter 9 through Appendix A.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101450487	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," Appendix B, Figure B-1 through Figure B-77.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101180508	APP-1200-S3R-003, Revision 2, "Design Report for the AP1000 Enhanced Shield Building," Appendix B, Figure B-1 through Figure B-77.		Westinghouse	DCP_NRC_002865
5/7/2010	ML101320314	APP-1200-S3R-003, Revision 2,"Design Report for the AP1000 Enhanced Shield Building," Appendix C, Section C.5 through Appendix G.		Westinghouse	DCP_NRC_002865
5/8/2010	ML101320318	06/9-11/2010 Notice of Public Meeting with Westinghouse to Discuss AP1000 Design Certification Amendment - Shield Building Design Methodology.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
5/10/2010	ML101320589	2010/05/10 AP-1000 DCD Review - OI-SRP9.1.3-SBPA-13 SFP AP1000 (1A)] for Review / Comments Due: 5/13/2010		Westinghouse	

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5/10/2010	ML101330044	2010/05/10 AP-1000 DCD Review - OI-SRP9.1.5-SBPB-01 AP1000 (2A)] for Review / Comments Due: 5/13/2010		Westinghouse	
5/10/2010	ML101330203	2010/05/10 AP-1000 DCD Review - Chapter 18 At Risk RAI Responses		- No Known Affiliation	
5/10/2010	ML101380249	2010/05/10 AP-1000 DCD Review - AP1000 P4 SER with No Open Items		NRC/NRO	
5/10/2010	ML101380250	2010/05/10 AP-1000 DCD Review - AP1000 SER - Section 11 Revision based upon staff review of response to RAI-SRP-11.3-CHPB-06		NRC/NRO	
5/10/2010	ML101380254	Description of Proposed Changes for AP1000 DCD Rev. 18.		Westinghouse	DCP_NRC_002863
5/10/2010	ML101400037	DCD Pages for CN59.		Westinghouse	DCP_NRC_002863
5/10/2010	ML101330142	Westinghouse Submittal of Information on Proposed Changes for AP1000 Design Certification Amendment Application, Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002863
5/10/2010	ML101340591	2010/05/10 AP-1000 DCD Review - RE: OI-SRP 9.1.3-SBPA-13 Rev. 1		- No Known Affiliation	
5/11/2010	ML101320319	AP1000 Response to Request for Additional Information (RAI-TR94-NSIR-20, Revision 1).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002871
5/12/2010	ML101320322	2010/05/12 AP-1000 DCD Review - drafts of OI-SRP9.1.3-SBPA-13 R1 and OI-SRP9.1.5-SBPB-01 R2		- No Known Affiliation	

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5/12/2010	ML101320323	2010/05/12 AP-1000 DCD Review - FW: OI-SRP9.1.4-SBPA-03 AP1000 (2A)] for Review / Comments Due: 5/13/2010		- No Known Affiliation	
5/12/2010	ML101320597	2010/05/12 AP-1000 DCD Review - RCP Variable Frequency drives.		- No Known Affiliation	
5/12/2010	ML101380347	2010/05/12 AP-1000 DCD Review - AP1000 Draft OI/RAI Responses - 5/12/10		NRC/NRO	
5/12/2010	ML101330047	2010/05/12 AP-1000 DCD Review - AP1000 Draft OI/RAI Responses - 5/12/10		NRC/NRO	
5/12/2010	ML101330204	Westinghouse Submittal of Revised AP100 Safeguards Assessment APP-GW-GLR-066, Revision 4.	Winters, J.	Westinghouse	NPP_NRC_000003
5/13/2010	ML101240969	2010/05/13 AP-1000 DCD Review - FW: AP1000 P4 SER with No Open Items		NRC/NRO	
5/13/2010	ML101330376	2010/05/13 AP-1000 DCD Review - FW: AP1000 SER - Section 11 Revision based upon staff review of response to RAI-SRP-11.3-CHPB-06		NRC/NRO	
5/14/2010	ML100910165	05/27/2010, Meeting With Westinghouse On AP1000 Design Certification Amendment - Nuclear Regulatory Commission Management Meeting On Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	

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5/14/2010	ML100910522	Transmittal of Safety Evaluation for AP1000 Design Control Document Amendment, Revision 17, Certification Phase 2.	McKirgan, J. B.	NRC/NRO/DSRA/ SPCV	
5/17/2010	ML101100194	Advanced Final Safety Evaluation Report For Chapter 10 Titled "Steam And Power Conversion System," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
5/17/2010	ML101250336	Steam And Power Conversion System.		NRC/NRO/DNRL/ NWE2	
5/18/2010	ML101370640	AP1000 Request for Withholding Information from Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of AP1000 Containment Debris Fuel Assembly Testing and Request for Additional Information Response Summary Presentation (DCP_NRC...	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002816 DPC_NRC_002812
5/18/2010	ML101400320	AP1000 Request for Withholding Information from Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of Design Report for the AP1000 Enhanced Shield Building (DCP_NRC_002830 and DCP_NRC_002835).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002830 DCP_NRC_002835
5/18/2010	ML100120788	Announcement Letter for NRC Design Certification Inspection of Westinghouse at Purdue University.	Peralta, J. D.	NRC/NRO/DCIP/CQVP	

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5/19/2010	ML100120839	2010/05/19 AP-1000 DCD Review - Acknowledgement of receipt of RAIs RAI-SRP3.7.1-SEB1-19, RAI-TR03-001 R1, -005 R2, -007 R2, and -037		- No Known Affiliation	
5/20/2010	ML100640742	Letter - Safety Evaluation Report With Open Items For Chapter 22, "Regulatory Treatment Of Non-Safety Systems," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793
5/20/2010	ML101200195	Safety Evaluation Report with Open Items For Chapter 22, "Regulatory Treatment Of Non-Safety Systems," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO	NUREG-1793
5/20/2010	ML101400038	P4 Chapter 4 (Reactor).		NRC/NRO/DNRL/ NWE2	
5/20/2010	ML101400249	Advanced Final Safety Evaluation Report for Chapter 4 Titled "Reactor," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
5/20/2010	ML101400322	2010/05/20 AP-1000 DCD Review - AP1000 DCD Capture - 5/20/10		NRC/NRO	
5/20/2010	ML101400571	Inspection Plan for Westinghouse AP1000 Design Certification - Dedication of Purdue Testing Services.	Jacobson, J.	NRC/NRO/DCIP/CQVB	

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5/20/2010	ML101400664	2010/05/20 AP-1000 DCD Review - FW: Acknowledgement of receipt of RAIs RAI-SRP3.7.1-SEB1-19, RAI-TR03-001 R1, -005 R2, -007 R2, and -037		NRC/NRO	
5/20/2010	ML101440386	Transmittal of Safety Evaluation for AP1000 Design Control Document Amendment, Revision 17, Certification Phase 2.	McKirgan, J. B. Terao, D. A.	NRC/NRO/DSRA/ SPCV	
5/20/2010	ML101440387	Transmittal of Safety Evaluation for AP1000 Design Control Document Amendment, Revision 17, Certification Phase 2.		NRC/NRO	
5/20/2010	ML101440388	Submittal of "AP1000 DCD Revision 18 Update Summary of New Change Notices - May 20, 2010," Presentation Slides.	Sisk, R. B.	Westinghouse	DCP_NRC_002884
5/20/2010	ML101450198	Submittal of AP1000 DCD Revision 18 Update Summary of New Change Notices - May 20, 2010 Presentation.		Westinghouse	DCP_NRC_002884
5/20/2010	ML101450300	Submittal of AP1000 DCD Revision 18 Update Summary of New Change Notices - May 20, 2010 Presentation.		Westinghouse	DCP_NRC_002884
5/20/2010	ML101300682	Westinghouse - Response to Proposed Open Item (Chapter 3), OI-SRP3.2.1-EMB2-06 R1, in Support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002886
5/20/2010	ML101300701	2010/05/20 AP-1000 DCD Review - RE: Draft RAI-TR09-08 Rev 5 to WEC		- No Known Affiliation	

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5/21/2010	ML101450248	Letter re: Advanced Final Safety Evaluation Report for Chapter 12 Titled "Radiation Protection," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
5/21/2010	ML101470257	SER - Advanced Final Safety Evaluation Report for Chapter 12 Titled "Radiation Protection," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO	NUREG-1793, Suppl 2
5/21/2010	ML101470258	Westinghouse, AP 1000 Response to Request for Additional Information (SRP 18), in support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002883
5/21/2010	ML101580471	Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	AW-10-2817 DCP_NRC_002874
5/21/2010	ML101450280	Enclosures 3, 6, Description of Proposed Changes for AP1000 DCD Rev. 18, Proprietary.		Westinghouse	DCP_NRC_002874
5/21/2010	ML101410193	AP1000 DCD Impact Document Submittal of APP-GW-GLR-115 (TR 115), Revision 2.	DeBlasio, J. Lapay, W. S. Sisk, R. B.	Westinghouse	AP1000 APP-GW-GLR-115, Rev 2 DCP_NRC_002889 TR 115
5/22/2010	ML092650374	2010/05/22 AP-1000 DCD Review - Draft RAI-SRP3.8.3-SEB1-04, Rev. 2		NRC/NRO	

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5/24/2010	ML101340177	Federal Register Notice Regarding the ACRS Subcommittee on AP1000, June 24-25, 2010.	Wang, W.	NRC/ACRS	
5/25/2010	ML101400117	P4 Chapter 11 (Radioactive Waste Management).	Sanders, S.	NRC/NRO/DNRL/ NWE2	
5/25/2010	ML101410466	Ltr - Advanced Final Safety Evaluation Report for Chapter 11 Titled "Radioactive Waste Management," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
5/25/2010	ML101450281	G20100259/EDATS: OEDO-2010-0329 - Linda Herr and Carolyn Harves, E-mail re: Briefing Package for Meeting with Westinghouse Electric on June 4, 2010.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	G20100259/EDATS: OEDO-2010-0329
5/25/2010	ML101450302	Advisory Committee On Reactor Safeguards AP1000 Subcommittee Review of Selected Chapters of the Safety Evaluation Report with Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
5/25/2010	ML101460555	2010/05/25 AP-1000 DCD Review - FW: Draft RAI-SRP3.8.3-SEB1-04, Rev. 2		NRC/NRO	
5/25/2010	ML101470307	2010/05/25 AP-1000 DCD Review - FW: Draft RAI-TR09-08 Rev 5 to WEC		NRC/NRO	
5/25/2010	ML101470308	Regulatory Audit Report for the AP1000 Sump Screen Downstream Effects/Long-Term Cooling Analyses.	Donoghue, J. E.	NRC/NRO/DSRA/SRS B	

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5/25/2010	ML101170634	Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002879
5/25/2010	ML101450006	Enclosure 3, Description of Proposed Changes for AP1000 DCD Rev. 18, Proprietary.		Westinghouse	DCP-NRC-002879 DCP_NRC_002818
5/26/2010	ML101450314	Audit Plan For Review of the AP1000 Fuel Rack Seismic Issues - June 2 - 3, 2010.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
5/26/2010	ML101450327	05/07/2010 Summary of the Audit Conducted on AP1000 Document APP-PXS-M3C-052, R1, at Westinghouse, Rockville, Maryland.	Makar, G. L.	NRC/NRO/DE/CIB1	APP-PXS-M3C-052, Rev 1
5/26/2010	ML101450331	04/26/2010-Summary of Category 1 Public Meeting Held with Westinghouse to Discuss AP1000 DC Adequacy of Seismic Margin.	Sanders, S. R.	NRC/NRO/DNRL/ NWE2	
5/26/2010	ML101450341	Westinghouse Draft AP1000 DCD SER Open Item Review Resolution.		Westinghouse	OI-SRP19.0-SPLA-12
5/26/2010	ML101480032	Westinghouse Draft AP1000 DCD SER Open Item Review Resolution.		Westinghouse	OI-SRP19.0-SPLA-12-01
5/26/2010	ML101480034	Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors DC/COL-ISG-020.		NRC/NRO/DNRL/ NWE2	DC/COL-ISG-020
5/26/2010	ML101520167	2010/05/26 AP-1000 DCD Review - AP1000 - New Draft RAI - RAI-SRP6.2.2-SPCV-25 R2		NRC/NRO	

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5/26/2010	ML101870162	2010/05/26 AP-1000 DCD Review - FW: AP1000 - New Draft RAI - RAI-SRP6.1.1-CIB1-02		NRC/NRO	
5/26/2010	ML101870164	2010/05/26 AP-1000 DCD Review - RE: AP1000 - New Draft RAI - RAI-SRP6.1.1-CIB1-02		- No Known Affiliation	
5/26/2010	ML101870168	2010/05/26 AP-1000 DCD Review - Acknowledgement of RAI-TR03-32 R5, and RAI-SRP3.7.1-SEB1-04 R4, -06 R5, and -15 R3		- No Known Affiliation	
5/26/2010	ML101470631	2010/05/26 AP-1000 DCD Review - Acknowledgement of Draft RAI-TR85-SEB1-36 R3		- No Known Affiliation	
5/26/2010	ML101470970	2010/05/26 AP-1000 DCD Review - Acknowledgement of OI-SRP7.1-ICE-02 R1, -03 R1; OI-SRP7.2-ICE-01 R1, -02 R1; OI-SRP7.8-DAS-04 R1 & OI-SRP7.9-ICE-04 R1		- No Known Affiliation	
5/27/2010	ML101470989	Advisory Committee on Reactor Safeguards AP1000 Subcommittee Review of Chapter 11, "Radioactive Waste Management," of The Safety Evaluation Report With Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
5/27/2010	ML101520023	Transmittal of Safety Evaluation for AP1000 DCD Amendment, Revision 17, Certification Phase 2 - Long Term Cooling.	Donoghue, J. E.	NRC/NRO/DSRA/SRSB	

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5/27/2010	ML101520025	Structural Engineering Branch 1 Input to Safety Evaluation Report with Open Items Under Standard Review Plan 3.8 for AP1000 Design Control Document, Revision 17.	Thomas, B. E.	NRC/NRO/DE/SEB1	
5/27/2010	ML101520188	2010/05/27 AP-1000 DCD Review - RE: AP1000 SER3.5.1.4 Supplemental		NRC/NRO	
5/27/2010	ML101530049	2010/05/27 AP-1000 DCD Review - RE: AP1000 Ammendment 17 SER for 3.2		NRC/NRO	
5/27/2010	ML101530050	AP1000 Response to Request for Additional Information (SRP3), RAI COL03.05.01.04-1 R2, in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002894
5/27/2010	ML101440374	AP1000 Response to Request for Open Item (SRP 5), OI-SRP5.4.1-CIB1-01 R1 in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002896 WEM_DCP_000472
5/28/2010	ML101480035	AP1000, Request for Withholding Information From Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of Design Report for the AP1000 Enhanced Shield Building (DCP_NRC_002830 and DCP_NRC_002835).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002830 DCP_NRC_002835
5/28/2010	ML101530046	2010/05/28 AP-1000 DCD Review - FW: AP1000 - New Draft RAI - RAI-SRP6.2.2-SPCV-25 R2		NRC/NRO	

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5/28/2010	ML101530048	2010/05/28 AP-1000 DCD Review - FW: AP1000 - New Draft RAI - RAI-SRP6.1.1-CIB1-02		NRC/NRO	
5/28/2010	ML101530052	Transmittal of Revision 4 to WCAP-15775, "AP1000 Instrumentation & Control Defense-in-Depth & Diversity Report."	Sisk, R. B.	Westinghouse	APP-GW-J1R-004 DCP_NRC_002901
5/28/2010	ML101530053	AP1000 Response to Request for Additional Information (SRP6.2.2).	Sisk, R. B.	Westinghouse	AP1000 AW-10-2826 DCP_NRC_002899 SRP 6.2.2
5/28/2010	ML101530055	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-133, Revision 1 (TR 133).	Sisk, R. B.	Westinghouse	DCP_NRC_002898
5/28/2010	ML101530056	AP1000 Response to Request for Additional Information (SRP 2), RAI-SRP2.2-RSAC-01 R1 and RAI-SRP2.5-RGS1-21 R1 in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002897
5/28/2010	ML101530183	Submittal of WCAP-17226, Revision 1, "Assessment of Potential Interactions Between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000 In-Core Instrumentation System."	Sisk, R. B.	Westinghouse	DCP_NRC_002903

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5/28/2010	ML101530184	WCAP-17226, Revision 1, "Assessment of Potential Interactions Between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000 In-Core Instrumentation System."	Ewald, J. G. Heibel, M. D.	Westinghouse	DCP_NRC _002903
5/28/2010	ML101540159	Submittal of WCAP-17179, Revision 1 - Proprietary "Component Interface Module Technical Report."	Sisk, R. B.	Westinghouse	AW-10-2829 DCP_NRC_002904 WCAP-17179, Rev 1
5/28/2010	ML101540160	WCAP-17179-P, Rev 1, "AP1000 Component Interface Module Technical Report."	Tweedle, T. W.	Westinghouse	APP-GW-GLR-143, Rev 1 DCP_NRC_002904
5/28/2010	ML101520021	Submittal of WCAP-17184, Revision 1 - Prop "AP1000TM Diverse Actuation System Planning and Functional Design Summary Technical Report".	Sisk, R. B.	Westinghouse	AP1000 AW-10-2827 DCP_NRC_002902
5/30/2010	ML101650116	2010/05/30 AP-1000 DCD Review - RE: Emailing: RAI-SRP7.0-ICE-01 thru RAI-SRP7.0-ICE-15 pkg 05-28-10.zip		NRC/NRO	
5/31/2010	ML101650117	APP-OCS-GEH-322 Revision D, "Human Factors Engineering Integrated System Validation Plan" (Non-Proprietary).	Reed, J. I.	Westinghouse	DCP_NRC_002912
5/31/2010	ML101650118	APP-OCS-GEH-323 Revision B, "Human Factors Engineering Integrated System Validation Scenario Information" (Non-Proprietary).	Briggs, D. W.	Westinghouse	DCP_NRC_002912

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5/31/2010	ML101520022	APP-OCS-GEH-320 Revision D, "Human Factors Engineering Integrated System Validation Plan" (Proprietary).	Reed, J. I.	Westinghouse	DCP_NRC_002912
5/31/2010	ML101520024	APP-OCS-GEH-321, Revision B, "Human Factors Engineering Integrated System Validation Scenario Information" (Proprietary).	Briggs, D. W.	Westinghouse	DCP_NRC_002912
6/1/2010	ML101520026	2010/06/01 AP-1000 DCD Review - FW: Emailing: RAI-SRP7.0-ICE-01 thru RAI-SRP7.0-ICE-15 pkg 05-28-10.zip		NRC/NRO	
6/1/2010	ML101520168	2010/06/01 AP-1000 DCD Review - FW: AP1000 SER3.5.1.4 Supplemental		NRC/NRO	
6/1/2010	ML101520201	2010/06/01 AP-1000 DCD Review - FW: AP1000 Ammendment 17 SER for 3.2		NRC/NRO	
6/1/2010	ML101520202	2010/06/01 AP-1000 DCD Review - FW: AP1000 - New Draft RAI - RAI-SRP6.1.1-CIB1-02		NRC/NRO	
6/1/2010	ML101520442	2010/06/01 AP-1000 DCD Review - RE: AP1000 - New Draft RAI's - RAI-SRP6.2.2-SRSB-43 and -44		- No Known Affiliation	
6/1/2010	ML101520444	2010/06/01 AP-1000 DCD Review - FW: AP1000 - New Draft RAI's - RAI-SRP6.2.2-SRSB-43 and -44		NRC/NRO	
6/1/2010	ML101520445	2010/06/01 AP-1000 DCD Review - Acknowledgement of OI-SRP 9.1.3-SBPA-13 R2		- No Known Affiliation	

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6/1/2010	ML101540246	2010/06/01 AP-1000 DCD Review - AP1000 DCD Capture - 6/1/2010		NRC/NRO	
6/1/2010	ML101550065	2010/06/01 AP-1000 DCD Review - FW: Acknowledgement of OI-SRP 9.1.3-SBPA-13 R2		NRC/NRO	
6/1/2010	ML101530009	2010/06/01 AP-1000 DCD Review - Acknowledgement of RAI-SRP3.12-EMB1-09		- No Known Affiliation	
6/1/2010	ML101530216	AP1000 Response to Request for Additional Information (SRP 6), RAI-SRP6.4-SPCV-16, in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_002907
6/2/2010	ML101530217	2010/06/02 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP3.12-EMB1-09		NRC/NRO	
6/2/2010	ML101530514	2010/06/02 AP-1000 DCD Review - Closure/dispositioning of two OIs for EMB (RAIs SRP3.9.1-EMB1-03 and 04		NRC/NRO	
6/2/2010	ML101530529	2010/06/02 AP-1000 DCD Review - FW: Closure/dispositioning of two OIs for EMB (RAIs SRP3.9.1-EMB1-03 and 04		NRC/NRO	
6/2/2010	ML101530531	06/15/10 Notice of Meeting with Westinghouse to Discuss AP1000 Instrumentation and Control System Open Issues to Complete Final Safety Evaluation Report.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	

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6/2/2010	ML101530582	2010/06/02 AP-1000 DCD Review - AP1000 Section 3.2.1 and 2 SE		NRC/NRO	
6/2/2010	ML101530583	2010/06/02 AP-1000 DCD Review - FW: AP1000 Section 3.2.1 and 2 SE		NRC/NRO	
6/2/2010	ML101540014	2010/06/02 AP-1000 DCD Review - Conversion of OI SRP3.9.3-EMB2-05 to a Confirmatory item		NRC/NRO	
6/2/2010	ML101580469	2010/06/02 AP-1000 DCD Review - FW: Conversion of OI SRP3.9.3-EMB2-05 to a Confirmatory item		NRC/NRO	
6/2/2010	ML101580470	2010/06/02 AP-1000 DCD Review -		NRC/NRO	
6/2/2010	ML101580475	Submittal of Non-Proprietary Document - WCAP-17184-NP, Revision 1, "AP1000 Diverse Actuation System Planning and Functional Design Summary Technical Report".	Darr, D. G. Peasley, S. A. Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002908 WCAP-17184-NP, Rev 1
6/2/2010	ML101540015	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-026, Revision 3 (TR44).	Sisk, R. B.	Westinghouse	APP-GW-GLR-026, Rev 3 DCP_NRC_002905 TR 44
6/2/2010	ML101540247	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-033, Revision 4 (TR54).	Sisk, R. B.	Westinghouse	DCP_NRC_002906
6/3/2010	ML101550678	2010/06/03 AP-1000 DCD Review - FW:		NRC/NRO	
6/3/2010	ML101520626	2010/06/03 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP3.12-EMB1-09		NRC/NRO	

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6/3/2010	ML101550137	2010/06/03 AP-1000 DCD Review - FW: Revised Responses to RAI SRP5.2.3-CIB1-01 Rev 2 Resulting from W/NRC Phone Call on May 26		- No Known Affiliation	
6/4/2010	ML101550168	AP1000, Request for Withholding Information from Public Disclosure, AP1000 Response to Open Item (OI-SRP8.3.2-Eeb-05) Proprietary and Non-Proprietary (DCP_NRC_002858).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002858
6/4/2010	ML101550677	06/09-11/2010 Public Meeting with Westinghouse on AP1000 Design Certification Amendment - Shield Building Design Methodology.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
6/4/2010	ML101550679	06/10/2010, 06/17/2010, & 06/24/2010 Notice of Meetings With Westinghouse On AP1000 Design Certification Amendment - Nuclear Regulatory Commission Meeting On Selected Technical Or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
6/4/2010	ML101540490	2010/06/04 AP-1000 DCD Review - RE: Draft of OI-SRP9.1.3-SBPA-13 R2.doc		- No Known Affiliation	
6/4/2010	ML101590167	2010/06/04 AP-1000 DCD Review - AP1000 DCD Capture - 6/4/10 Drafts		NRC/NRO	

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6/7/2010	ML101670075	Transmittal of Safety Evaluation for AP1000 Design Control Document Amendment, Revision 17, Certification Phase 2.	McKirgan, J. B.	NRC/NRO/DSRA/ SBCV	AP1000
6/7/2010	ML101590083	2010/06/07 AP-1000 DCD Review - SUBMITTAL4BC - Final Supplement Input from SBPB to AP1000 DCD Safety Evaluation Report Section 3.5.1.4		NRC/NRO	
6/7/2010	ML101590164	2010/06/07 AP-1000 DCD Review - RE: Seismic Margin RAI - Chapter 19		- No Known Affiliation	
6/8/2010	ML101590168	2010/06/08 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP3.12-EMB1-09		NRC/NRO	
6/8/2010	ML101590496	2010/06/08 AP-1000 DCD Review - FW: SEB1 Contribution to AP1000 DCD SER 3.5.1.4		NRC/NRO	
6/8/2010	ML101590497	2010/06/08 AP-1000 DCD Review - FW: SUBMITTAL4BC - Final Supplement Input from SBPB to AP1000 DCD Safety Evaluation Report Section 3.5.1.4		NRC/NRO	
6/8/2010	ML101650114	2010/06/08 AP-1000 DCD Review - FW: SUBMITTAL4BC - Final Supplement Input from SBPB to AP1000 DCD Safety Evaluation Report Section 3.5.1.4		NRC/NRO	

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6/8/2010	ML101520592	2010/06/08 AP-1000 DCD Review - FW: SUBMITTAL4BC - Final Supplement Input from SBPB to AP1000 DCD Safety Evaluation Report Section 3.5.1.4		NRC/NRO	
6/8/2010	ML101540188	Submittal of Proprietary and Non-Proprietary Integrated System Validation (ISV) Documents for Docketing.	Sisk, R. B.	Westinghouse	AW-10-2833 DCP_NRC_002912
6/9/2010	ML101600521	AP1000 Request for Withholding Information from Public Disclosure, Transmittal Of Proprietary and Non-Proprietary Versions Of Design Report for the AP1000 Enhanced Shield Building (DCP_NRC_002830 and DCP_NRC_002835).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002830 DCP_NRC_002835
6/9/2010	ML101600522	05/26/2010-Summary of Category 2 Public Meeting with Nuclear Energy Institute to Discuss Health Physics Issues for New Reactors.	Stutzcage, E. E.	NRC/NRO/DCIP/CHPB	
6/9/2010	ML101600523	2010/06/09 AP-1000 DCD Review - Additional information relative to RAI SRP5.2.3-CIB1-01 Rev 2		- No Known Affiliation	
6/9/2010	ML101650261	2010/06/09 AP-1000 DCD Review - FW: RAI-SRP5.4.1 CQVB-01 Draft R!		- No Known Affiliation	
6/9/2010	ML101650263	2010/06/09 AP-1000 DCD Review - AP1000 DCD Capture - 6/9/10 draft RAIs responses		NRC/NRO	

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6/9/2010	ML101760088	WESTEMS User's Manual Addenda: Guidance Documents for the User.	Sisk, R. B.	Westinghouse	DCP_NRC_002914
6/9/2010	ML101610444	"WESTEMS Version 4.5.2 User's Manual Addendum 3: Peak and Valley Selection and Documentation Guidelines," Revision 0, dated May 28, 2010.		Westinghouse	DCP_NRC_002914 LTR-PAFM-10-100
6/9/2010	ML101610459	Westinghouse AP1000 Shield Building Structural Review Update - Public Session, June 9, 2010 by Bruce Bevilacqua, Vice President, Engineering, Westinghouse Nuclear Power Plants	Bevilacqua, B.	Westinghouse	
6/10/2010	ML101720019	Memo: Safety Evaluation Report (Ser) For The AP1000 DCD FSAR (Proposed To Revision 17) Section 2.2 Nearby Industrial, Transportation, And Military Facilities.	Samaddar, S. K.	NRC/NRO/DSER/RSAC	
6/10/2010	ML101620262	Enclosure: Safety Evaluation Report (SER) For The AP1000 DCD FSAR (Proposed To Revision 17) Section 2.2 Nearby Industrial, Transportation, And Military Facilities. Enclosure: Safety Evaluation Report (SER) for The AP1000 DCD Final Safety Analysis Report.		NRC/NRO/DSER/RSAC	

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6/10/2010	ML101720011	2010/06/10 AP-1000 DCD Review - RE: Revised Responses to RAI SRP5.2.3-CIB1-01 Rev 2 Resulting from W/NRC Phone Call on May 26		- No Known Affiliation	
6/11/2010	ML101650058	AP1000 Revision 15 Certified Design Complete ITAAC List.	Jardaneh, M. M.	NRC/NRO/DCIP/CTSB	
6/13/2010	ML101670133	2010/06/13 AP-1000 DCD Review - smaller file sizes		- No Known Affiliation	
6/14/2010	ML101670135	2010/06/14 AP-1000 DCD Review - FW: Seismic Margin RAI - Chapter 19		NRC/NRO	
6/14/2010	ML101670136	AP1000 Response to Request for Additional Information (SRP6.2.2), RAI-SRP6.2.2-SRSB-44 in Support of the Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AW-10-2853 DCP_NRC_002917
6/14/2010	ML101720013	Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	AW-10-2830 DCP_NRC_002909
6/14/2010	ML101540211	2010/06/14 AP-1000 DCD Review - Fuel Rack RAI Drafts (per audit discussions) - audit actions 8 and 16		- No Known Affiliation	
6/15/2010	ML101670077	Follow-up Audit of Design and Procurement Specifications for Pumps, Valves, and Dynamic Restraints for AP1000 Reactor.	Scarbrough, T. G.	NRC/NRO/DE/CIB2	

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6/16/2010	ML101680654	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of The AP1000 Calorimetric Power Uncertainty April 26, 2010, Presentation (DCP_NRC_002857).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002857
6/16/2010	ML101720014	2010/06/16 AP-1000 DCD Review - AP1000 DCD Capture - 6/16/10		NRC/NRO	
6/16/2010	ML101690454	6/16/2010 Construction Reactor Oversight Process Category 2 Public Meeting Handout: Risk Importance Table.		NRC/NRO/DCIP	
6/16/2010	ML101690455	2010/06/16 AP-1000 DCD Review - Resubmit proper draft RAI-TR44-09 R2A		- No Known Affiliation	
6/17/2010	ML101820084	Chapter 9 Safety Evaluation Report (SER) with Open Items Proprietary Review - Update.	Sisk, R. B.	Westinghouse	DCP_NRC_002919
6/17/2010	ML101720017	Enclosure 2, Westinghouse AP1000 Chapter 9 SER - Update List of Proprietary Information and Text.		Westinghouse	DCP_NRC_002919
6/17/2010	ML101720018	2010/06/17 AP-1000 DCD Review - Response to RAI-SRP6 1 1-CIB1-02 R0-draft.pdf		- No Known Affiliation	
6/18/2010	ML101720020	2010/06/18 AP-1000 DCD Review - Draft version of RAI-TR44-01 R1B		- No Known Affiliation	
6/18/2010	ML101720645	2010/06/18 AP-1000 DCD Review - AP1000 DCD Capture - 6/18/10 drafts		NRC/NRO	

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6/18/2010	ML101720646	2010/06/18 AP-1000 DCD Review - AP1000 DCD Capture - 6/18/10 drafts		NRC/NRO	
6/18/2010	ML101380217	Submittal of Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002918
6/21/2010	ML101680069	Audit Report For Lifecycle Phases One And Two Documentation Validation Related To Protection And Safety Monitoring System And Diverse Actuation System For The Ap1000 Design Certification Amendment Application	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/21/2010	ML101740602	Audit Report For Lifecycle Phases One And Two Documentation Validation Related To Protection And Safety Monitoring System And Diverse Actuation System For The AP1000 Design Certification Amendment Application.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/21/2010	ML101790090	Ltr to Sadler Rupprecht from David Matthews re: AP1000 Design Certification Amendment Schedule.	Matthews, D. B.	NRC/NRO/DNRL	
6/21/2010	ML101790091	Letter - Audit Report For Lifecycle Phases One And Two Documentation Validation Related To Protection And Safety Monitoring System And Diverse Actuation System For The AP1000 Design Certification Amendment Application.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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6/22/2010	ML101790362	AP1000 Response to Request for Additional Information (SRP6.2.2).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002926 SRP6.2.2
6/22/2010	ML101590617	Westinghouse Submittal of Chapter 7 Documents.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002927
6/23/2010	ML101720170	IR 05200006-10-201, on 05/11 - 13, 2010, Westinghouse Facility in Cranberry Township, PA, Assessment of Quality Activities Implemented to Control Use of a Macro Code Used in the Design of Nuclear Island Structures of AP1000 Design.	Peralta, J. D.	NRC/NRO/DCIP/CQVP	
6/23/2010	ML101720236	05/20/2010 Summary of Category 1 Public Meeting With Westinghouse to Discuss Design Changes to be Incorporated in Revision 18 to the AP1000 Design Control Document.	Proctor, C. M.	NRC/NRO/DNRL/ NWE2	
6/23/2010	ML101790088	Safety Evaluation Report Section 3.9.6 on Revision 17 to Westinghouse AP1000 Design Certification Amendment.	Terao, D.	NRC/NRO/DE/CIB1	
6/23/2010	ML101790089	Mechanical Equipment Environmental Qualification Input for Safety Evaluation Report Section 3.11 on Revision 17 to Westinghouse AP1000 Design Certification Amendment.	Terao, D.	NRC/NRO/DE/CIB1	
6/24/2010	ML101660670	Final Shield Building In-Plane Shear Test Results.	Sisk, R. B.	Westinghouse	DCP_NRC_002928

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6/25/2010	ML101580228	Transmittal of Safety Evaluation for AP1000 Design Certification Document Amendment Revision 17, Certification Phase 2.	McKirgan, J. B.	NRC/NRO/DSRA/ SBCV	
6/25/2010	ML101600487	Transmittal of Safety Evaluation for AP1000 Design Control Document Amendment Revision 17, Certification Phase 2.		NRC/NRO/DSRA/ SBCV	
6/28/2010	ML101740222	Advanced Final Safety Evaluation Report For Chapter 16 Titled "Technical Specification," Of NUREG-1793, Supplement 2 - AP1000 Design Certification. Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/28/2010	ML101830084	SER - Advanced Final Safety Evaluation Report For Chapter 16 "Technical Specification," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.		NRC/NRO/DNRL/ NWE2	
6/28/2010	ML101410310	05/20/2010 Summary Of A Category 1 Public Meeting Held With Westinghouse In Rockville, Maryland Regarding Open Items In AP1000 Safety Evaluation Report.	Proctor, C. M.	NRC/NRO/DNRL/ NWE2	
6/28/2010	ML101410395	2010/06/28 AP-1000 DCD Review - Fuel rack Final Draft RAIs: RAI-TR44-017-3C, RAI-TR54-01 R1C		- No Known Affiliation	

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6/29/2010	ML101540139	Audit Report For The Review Of Proprietary Technical, Procedural, And Process Information Related To The Component Interface Module And Diverse Actuation System For The AP1000 Design Certification Amendment Application.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/29/2010	ML101540170	Audit Report For The Review Of Proprietary Technical, Procedural, And Process Information Related To The Component Interface Module And Diverse Actuation System For The AP1000 Design Certification Amendment Application.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
6/29/2010	ML101800374	Letter - Advanced Final Safety Evaluation Report For Chapter 2 Titled "Site Envelop," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
6/29/2010	ML101800387	SER - Advanced Final Safety Evaluation Report For Chapter 2 Titled "Site Envelop."		NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
6/29/2010	ML101800507	Audit Report For AP1000 Design Certification Probabilistic Risk Assessment - October 13-19, 2009.	Sanders, S.	NRC/NRO/DNRL/ NWE2	
6/29/2010	ML101800508	Audit Report For AP1000 Design Certification PRA.	Sanders, S.	NRC/NRO/DNRL/ NWE2	
6/29/2010	ML101830081	2010/06/29 AP-1000 DCD Review - Revised Draft of the response to RAI-SRP6 1 1-CIB1-02 R0		- No Known Affiliation	

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6/29/2010	ML101830085	2010/06/29 AP-1000 DCD Review - AP1000 DCD Capture - 6/29/10 draft		NRC/NRO	
6/29/2010	ML102030186	2010/06/29 AP-1000 DCD Review - RAI-SRP5 4 1-CQVB-01 Draft R1 6-29-10.doc		- No Known Affiliation	
6/29/2010	ML101730168	2010/06/29 AP-1000 DCD Review - New Fuel Rack RAIs - per Non-credible "drop" position		- No Known Affiliation	
6/29/2010	ML101810131	Transmittal of IRWST and CR Screen Related Documents.	Sisk, R. B.	Westinghouse	DCP_NRC_002931
6/30/2010	ML101820514	Safety Evaluation Report - SRP Section 8 for Revision to Westinghouse AP1000 DCD Design Control Document.	Jenkins, R. V.	NRC/NRO/DE/EEB	AP1000
6/30/2010	ML101830078	Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, July 21-22, 2010.	Wang, W.	NRC/ACRS	
6/30/2010	ML101830079	Summary List of ISG-11 Design Change Proposals for the AP1000 Design Control Document, Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002934
6/30/2010	ML101830396	2010/06/30 AP-1000 DCD Review - Acknowledgement of : RAI-DCP-CN58-SRSB-01		- No Known Affiliation	
6/30/2010	ML101870158	2010/06/30 AP-1000 DCD Review - Acknowledgement of : AP1000 RAI-DCP-CN8-SRSB-01		- No Known Affiliation	
6/30/2010	ML101940044	10 CFR 50.46 Thirty (30) Day Report for the AP1000 Standard Plant Design.	Sisk, R. B.	Westinghouse	DCP_NRC_002943
6/30/2010	ML101940046	2010/06/30 AP-1000 DCD Review - Acknowledgement of: RAI-SRP3.10-EMB-11		- No Known Affiliation	

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6/30/2010	ML101940048	Presentations and Actions for WEC Meeting with NRC June 9 through June 11.	Sisk, R. B.	Westinghouse	AW-10-2864 DCP_NRC_002929
6/30/2010	ML101550290	APP-GW-J0R-012, Revision 1, "AP1000 Protection and Safety Monitoring System Computer Security Plan."	Batson, S. J.	Westinghouse	
7/1/2010	ML101750759	04/08/10 Summary of Meeting with Westinghouse to Discuss Component Interface Module (CIM) And Diverse Actuation System (DAS) Technical Reports (TR) of the AP1000 Instrumentation and Control System.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
7/1/2010	ML101750762	PROBLEM AP1000 Diverse Actuation System (DAS) Setpoint Methodology – Westinghouse Responses to NRC Questions	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
7/1/2010	ML101750763	Phase 2 Safety Evaluation Report for Chapter 3 Section 3.7 Titled "Seismic Design" and Chapter 3 Section 3.8 Titled "Design of Category I Structures" of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment - Without Shield Building (Letter).	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
7/1/2010	ML101790099	Safety Evaluation Report - Chapter 3, Section 3.7, "Seismic Design."		NRC/NRO/DNRL/ NWE1	NUREG-1793, Suppl 2
7/1/2010	ML101820085	Safety Evaluation Report - Chapter 3, Section 3.8 "Design of Category I Structures."		NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2

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7/1/2010	ML101820483	07/01/2010 CIP Category 3 Public Meeting DAC Inspection Summary.	Fredette, T. R.	NRC/NRO/DCIP/CAEB	
7/1/2010	ML101830067	2010/07/01 AP-1000 DCD Review - AP1000 DCD Capture - 6.18.10 RAI response		NRC/NRO	
7/1/2010	ML101830069	07/01/2010 CIP Category 3 Public Meeting Surge in ITAAC Submittals Westinghouse Presentation.	Ray, T.	Westinghouse	
7/1/2010	ML101830070	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN08-ICE-02		- No Known Affiliation	
7/1/2010	ML101830071	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN65-ICE-01		- No Known Affiliation	
7/1/2010	ML101830072	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN64-ICE-01		- No Known Affiliation	
7/1/2010	ML101830073	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN63-ICE-01		- No Known Affiliation	
7/1/2010	ML101830074	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN60-ICE-02		- No Known Affiliation	
7/1/2010	ML101830075	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN66-SRSB-04		- No Known Affiliation	
7/1/2010	ML101830076	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN66-SRSB-03		- No Known Affiliation	
7/1/2010	ML101830077	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN66-SRSB-02		- No Known Affiliation	
7/1/2010	ML101680449	2010/07/01 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN66-SRSB-01		- No Known Affiliation	

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7/1/2010	ML101680683	2010/07/01 AP-1000 DCD Review - Acknowledgement of : AP1000 RAI-DCP-CN60-SRSB-01		- No Known Affiliation	
7/2/2010	ML101830066	Letter - Advanced Final Safety Evaluation Report For Chapter 17 Titled "Quality Assurance," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
7/2/2010	ML101830083	Safety Evaluation, Chapter 17 - Quality Assurance.		NRC/NRO/DNRL	NUREG-1793, Suppl 2
7/2/2010	ML101830086	2010/07/02 AP-1000 DCD Review - AP1000 DCD Capture - 7/2/10 drafts		NRC/NRO	
7/2/2010	ML101830136	2010/07/02 AP-1000 DCD Review - AP1000 DCD Capture - 7/2/10 2		NRC/NRO	
7/2/2010	ML101870123	2010/07/02 AP-1000 DCD Review - AP1000 DCD Capture - 7/2/10 3		NRC/NRO	
7/2/2010	ML101880536	Advisory Committee On Reactor Safeguards AP1000 Subcommittee Review Of Selected Chapters Of The Safety Evaluation Report With Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
7/2/2010	ML101890585	2010/07/02 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN59-EEB-02		- No Known Affiliation	
7/2/2010	ML101870113	2010/07/02 AP-1000 DCD Review - FW: Final Seismic OI draft version		- No Known Affiliation	
7/2/2010	ML101870114	AP1000 Response to Request for Open Item (SRP 3).	Sisk, R. B.	Westinghouse	DCP_NRC_002945

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7/6/2010	ML101870115	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN08-ICE-02		NRC/NRO	
7/6/2010	ML101870116	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN65-ICE-01		NRC/NRO	
7/6/2010	ML101870117	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN64-ICE-01		NRC/NRO	
7/6/2010	ML101870118	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN63-ICE-01		NRC/NRO	
7/6/2010	ML101870119	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN60-ICE-02		NRC/NRO	
7/6/2010	ML101870120	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN66-SRSB-04		NRC/NRO	
7/6/2010	ML101870121	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN66-SRSB-03		NRC/NRO	
7/6/2010	ML101870122	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN66-SRSB-02		NRC/NRO	
7/6/2010	ML101870124	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN66-SRSB-01		NRC/NRO	

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7/6/2010	ML101870154	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of : AP1000 RAI-DCP-CN8-SRSB-01		NRC/NRO	
7/6/2010	ML101870156	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN59-EEB-02		NRC/NRO	
7/6/2010	ML101870159	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of : AP1000 RAI-DCP-CN60-SRSB-01		NRC/NRO	
7/6/2010	ML101870160	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of : RAI-DCP-CN58-SRSB-01		NRC/NRO	
7/6/2010	ML101870163	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of: RAI-SRP3.10-EMB-11		NRC/NRO	
7/6/2010	ML101870166	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of OI-SRP 9.1.3-SBPA-13 R2		NRC/NRO	
7/6/2010	ML101870169	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-TR03-32 R5, and RAI-SRP3.7.1-SEB1-04 R4, -06 R5, and -15 R3		NRC/NRO	
7/6/2010	ML101870170	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of Draft RAI-TR85-SEB1-36 R3		NRC/NRO	

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7/6/2010	ML101870501	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of OI-SRP7.1-ICE-02 R1, -03 R1; OI-SRP7.2-ICE-01 R1, -02 R1; OI-SRP7.8-DAS-04 R1 & OI-SRP7.9-ICE-04 R1		NRC/NRO	
7/6/2010	ML101870502	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of receipt of RAIs RAI-SRP3.7.1-SEB1-19, RAI-TR03-001 R1, -005 R2, -007 R2, and -037		NRC/NRO	
7/6/2010	ML101870615	2010/07/06 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN66-CTSB-05		- No Known Affiliation	
7/6/2010	ML101870620	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN66-CTSB-05		NRC	
7/6/2010	ML101880515	2010/07/06 AP-1000 DCD Review - Acknowledgement of RAI-DCP-CN45-SBP-01		- No Known Affiliation	
7/6/2010	ML101880533	2010/07/06 AP-1000 DCD Review - FW: Acknowledgement of RAI-DCP-CN45-SBP-01		NRC	
7/6/2010	ML101880708	2010/07/06 AP-1000 DCD Review - OI-914-03 R3A		- No Known Affiliation	
7/6/2010	ML101880710	2010/07/06 AP-1000 DCD Review - ISI Potential Words		- No Known Affiliation	
7/6/2010	ML101880531	Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002925

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7/6/2010	ML101880537	Description of Proposed Changes for AP1000 DCD Rev. 18, Proprietary - Change No. 69, Direct Vessel Injection (DVI) Nozzle Design Change and Change No. 70, PCS Design Changes.		Westinghouse	DCP_NRC_002925
7/7/2010	ML101880727	2010/07/07 AP-1000 DCD Review - Fuel Rack RAIs		- No Known Affiliation	
7/7/2010	ML101880759	2010/07/07 AP-1000 DCD Review - AP1000 DCD Capture - 7/7/2010		NRC	
7/7/2010	ML101880762	2010/07/07 AP-1000 DCD Review - AP1000 DCD Capture - 7/7/2010 2		NRC	
7/7/2010	ML101880763	2010/07/07 AP-1000 DCD Review - AP1000 DCD Capture 7/7/2010 4		NRC	
7/7/2010	ML101870359	2010/07/07 AP-1000 DCD Review - FW: smaller files		NRC	
7/7/2010	ML101890439	2010/07/07 AP-1000 DCD Review - AP1000 DCD Capture 7/7/2010 3		NRC	
7/8/2010	ML101890546	06/10/2010 Summary of a Public Meeting to Discuss AP1000 Design Control Document Closure of Chapter 5 Issues.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
7/8/2010	ML101930129	2010/07/08 AP-1000 DCD Review - FW: 3.9.3 Audit Follow Up		NRC/NRO	
7/8/2010	ML101930130	Submittal of Certified Copies of Agreements.	Moore, S. W.	NRC/OIP	
7/8/2010	ML102030138	Submittal of Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002932

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7/8/2010	ML101930097	Description of Proposed Changes for AP1000 DCD, Revision 18.		Westinghouse	DCP_NRC_002932
7/8/2010	ML101930098	2010/07/08 AP-1000 DCD Review - RE: OI-914-03 R3A		- No Known Affiliation	
7/12/2010	ML101930144	2010/07/12 AP-1000 DCD Review - Acknowledgement of RAI-SRP19F-AIA-01 through -10 (Aircraft Impact RAIs)		- No Known Affiliation	
7/12/2010	ML101960079	2010/07/12 AP-1000 DCD Review - FW: Acknowledgement of RAI-SRP19F-AIA-01 through -10 (Aircraft Impact RAIs)		NRC/NRO	
7/12/2010	ML101930101	07/29, 08/05, 12 and 19/2010 Revised Notice of Meeting with Westinghouse on AP1000 Design Certification Amendment - Nuclear Regulatory Commission Meeting on Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
7/12/2010	ML101970030	2010/07/12 AP-1000 DCD Review - OI-911-08		- No Known Affiliation	
7/13/2010	ML101940162	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of Presentations And Actions For Westinghouse Company Meeting With U.S. Nuclear Regulatory Commission From June 9 Through June 11.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002929

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7/13/2010	ML101960078	AP1000 Submittal of APP-GW-GLE-002, Rev. 7, "Impacts to the AP1000 to Address Generic Safety Issue 191".	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_002962 GSI-191
7/14/2010	ML101960082	07/23/2010 Notice of Meeting with Westinghouse on AP1000 Containment External Pressure Analysis and Design.	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	
7/14/2010	ML102000154	2010/07/14 AP-1000 DCD Review - RAI-TR44-01 R1D (parallel with TR44-06 R2B-a)		- No Known Affiliation	
7/14/2010	ML102040724	2010/07/14 AP-1000 DCD Review - Eight Fuel Rack Audit Draft RAIs (with detail date info)		- No Known Affiliation	
7/14/2010	ML101960084	Letter re: Resubmittal of Proprietary and Non-Proprietary Versions of WCAP-16914, Rev. 0, "Evaluation of Debris Loading Head Loss Tests for AP 1000 Recirculation Screens and In-Containment Refueling Water Storage Tank Screens".	Sisk, R. B.	Westinghouse	DCP_NRC_002961
7/14/2010	ML102000096	2010/07/14 AP-1000 DCD Review - Gas Accumulation draft DCD Changes		- No Known Affiliation	
7/15/2010	ML102000097	2010/07/15 AP-1000 DCD Review - AP1000 DCD Capture - 7/15/10		NRC/NRO	
7/15/2010	ML102000098	Submittal of "AP1000 DCD Revision 18 - Closure of Recent Design Changes - July 15, 2010" Presentation Material.	Sisk, R. B.	Westinghouse	DCP_NRC_002947

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7/16/2010	ML102010269	IR 05200006-10-202 on 05/25/10 - 05/28/10 and Notice of Violation for Westinghouse.	Peralta, J. D.	NRC/NRO/DCIP/CQVB	
7/16/2010	ML101730094	2010/07/16 AP-1000 DCD Review - 7/13/2010 - Summary of Phone Call with Westinghouse to Gain Clarification on Draft RAI Response RAI-DCP-CN59-SBP-01		NRC/NRO	
7/16/2010	ML102030123	Westinghouse Submittal of AP1000 Safeguards Assessment APP-GW-GLR-066, Revision 5.	Winters, J.	Westinghouse	DCP_NRC_002969
7/19/2010	ML102030134	06/09-11/2010-Summary of Category 1 Public Meeting Held with Westinghouse Regarding the Proposed AP1000 Shield Building Design Methodology, Held in Rockville, Maryland.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	
7/20/2010	ML102170123	2010/07/20 AP-1000 DCD Review - RE: Draft Rev 2 of RAI-SRP5 2 3-CIB1-03 (Quickloc)		NRC	
7/20/2010	ML102170124	2010/07/20 AP-1000 DCD Review - RE: Chapter 19 Open Item on Seismic Margin		NRC	
7/20/2010	ML101820356	Transmittal of Technical Report APP-GW-GLR-079 Revision 8 (TR-026), "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."	Sisk, R. B.	Westinghouse	APP-GW-GLR-079, Rev 1 DCP_NRC_002968 TR-026

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7/22/2010	ML102030136	Summary of A Category 1 Public Meeting Held With Westinghouse Regarding The Proposed AP1000 Design Change Packages, Held In Rockville, Maryland On May 20, 2010.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
7/22/2010	ML102030139	AP1000 DCD Revision 18 Update Summary of New Change Notices May 20, 2010.		Westinghouse	
7/22/2010	ML102040722	2010/07/22 AP-1000 DCD Review -		- No Known Affiliation	
7/22/2010	ML102040723	2010/07/22 AP-1000 DCD Review - AP1000 DCD Capture - 7/22/10		NRC	
7/23/2010	ML102040725	2010/07/23 AP-1000 DCD Review - Draft of RAI-SRP18-COLP-46 R2A - Draft.doc		- No Known Affiliation	
7/23/2010	ML102100225	2010/07/23 AP-1000 DCD Review - FW: Draft of RAI-SRP18-COLP-46 R2A - Draft.doc		NRC	
7/23/2010	ML101890155	2010/07/23 AP-1000 DCD Review - FW: Gas Accumulation draft DCD Changes		NRC	
7/23/2010	ML101890223	2010/07/23 AP-1000 DCD Review - RE: AP1000 RAI-DCP-CN58-SRSB-02		- No Known Affiliation	
7/28/2010	ML102010738	Advanced Final Safety Evaluation Report for Chapter 8 Titled "Electric Power Systems," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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7/28/2010	ML102080426	Chapter 8, Eclectic Power Systems, Advanced Final Safety Evaluation Report.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
7/28/2010	ML102110180	Audit Report for AP1000 DCA Fuel Rack Seismic Issues - June 2 - 4, 2010.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
7/28/2010	ML102100226	08/26/2010 and 09/02, 09, & 16/2010, Revised Notice of Meeting with Westinghouse on AP1000 Design Certification Amendment - on Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
7/28/2010	ML102140333	Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002939
7/29/2010	ML102140334	2010/07/29 AP-1000 DCD Review - AP1000 DCD Capture - 7/29/10		NRC	
7/29/2010	ML102040199	Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002940
7/29/2010	ML102160220	Description of Proposed Changes for AP1000 DCD Rev. 18, Proprietary.		Westinghouse	DCP_NRC_002940
7/30/2010	ML102170267	AP1000 Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions Final Shield Building In-Plane Shear Test Results (DCP_NRC_002928).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002928

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7/30/2010	ML102160221	Transmittal of Technical Report APP-GW-GLR-096 Revision 0, (Proprietary) & APP-GW-GLR-097 Revision 0 (Non-Proprietary) "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses".	Sisk, R. B.	Westinghouse	APP-GW-GLR-096 APP-GW-GLR-097 DCP_NRC_002986
7/30/2010	ML102160222	APP-GW-GLR-145, WCAP-17184-P, Revision 2, "AP1000 Diverse Actuation System Planning and Functional Design Summary Technical Report."	Peasley, S. A.	Westinghouse	
7/31/2010	ML102170038	APP-GW-GLR-097, Revision 0, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the containment Response & Response and Safety Analyses."		Westinghouse	DCP_NRC_002986
7/31/2010	ML102170039	APP-GW-GLR-096, Revision 0, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses."		Westinghouse	DCP_NRC_002986
7/31/2010	ML102170259	APP-OCS-GEH-521 Rev. B, "AP1000 Plant Startup Human Factors Engineering Design Verification Plan."	Ma, R.	Westinghouse	DCP_NRC_002990
7/31/2010	ML102170260	APP-OCS-GEH-520, Rev. B, "AP1000 Plant Startup Human Factors Engineering Design Verification Plan."	Ma, R.	Westinghouse	DCP_NRC_002990

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7/31/2010	ML102170263	WCAP-17179-NP, APP-GW-GLR-144, Revision 2, "AP1000 Component Interface Module Technical Report."	Tweedle, T. W.	Westinghouse	APP-GW-GLR-144, Rev 2 DCP_NRC_002991
7/31/2010	ML102170264	WCAP-17226-NP, Revision 1, "Assessment of Potential Interactions Between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000 In-Core Instrumentation System."	Heibel, M. D.	Westinghouse	
7/31/2010	ML102170265	APP-GW-GLR-146, WCAP-17184-NP, Revision 2, "AP1000 Diverse Actuation System Planning and Functional Design Summary Technical Report."	Peasley, S. A.	Westinghouse	APP-GW-GLR-146
7/31/2010	ML102170266	APP-GW-J0R-013, Rev. 1, AP1000 Protection and Safety Monitoring System Computer Security Plan.		Westinghouse	
7/31/2010	ML102040143	WCAP-17179-P, APP-GW-GLR-143, Revision 2, "AP1000 Component Interface Module Technical Report."	Tweedle, T. W.	Westinghouse	
7/31/2010	ML102170037	WCAP-17226-P, Revision 1, "Assessment of Potential Interactions Between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000 In-Core Instrumentations System."	Heibel, M. D.	Westinghouse	

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8/2/2010	ML102170258	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Information On Proposed Changes For The AP1000 Design Control Document Revision 18 (DCP_NRC_002909).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002909
8/2/2010	ML102170034	Submittal of the AP1000 Plant Startup Human Factors Engineering Design Verification Plan; Response to RAI-SRP18-COLP-23; and Response to RAI-SRP18-COLP-46.	Sisk, R. B.	Westinghouse	AW-10-2898 DCP_NRC_002990
8/2/2010	ML102420489	Submittal of AP1000 Instrumentation and Control Documents to Support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AW-10-2899 DPC_NRC_002991
8/3/2010	ML102420490	Westinghouse Submittal of APP-GW-GLR-026, Revision 4 of AP1000 Standard Combined License Technical Report No. 44 (TR44), "New Fuel Storage Rack Structural/Seismic Analysis."	Sisk, R. B.	Westinghouse	DCP_NRC_002999
8/3/2010	ML102420503	Presentations & Actions for WEC Meeting With NRC on June 9 through June 11, 2010 re Design Report for AP1000 Enhanced Shield Building.	Sisk, R. B.	Westinghouse	DCP_NRC_002995
8/3/2010	ML102210126	Presentations & Actions for WEC Meeting With NRC on June 9 through June 11, 2010 re Shear Friction Behavior.	Orr, R.	Westinghouse	DCP_NRC_002995

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8/3/2010	ML102150554	Presentations & Actions for WEC Meeting With NRC on June 9 through June 11, 2010 re Response to NRC Action Items.		Westinghouse	DCP_NRC_002995
8/4/2010	ML102210127	Westinghouse - Response to Request for Additional Information (SRP 23), RAI-DCP-CN58-SRSB-01, in support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_003000 SRP 23
8/5/2010	ML102180040	Advance Safety Evaluation Report Input for Chapter 7, Instrumentation and Controls, for Westinghouse AP1000 Design Control Documentation, Revision 17 (TAC#RB7500).	Jackson, T. W.	NRC/NRO/DE/ICE1	TAC RB7500
8/5/2010	ML102220579	Westinghouse - Response to Request for Additional Information (SRP 23), RAI-DCP-CN59-EEB-02, in support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_003002 SRP 23
8/6/2010	ML102220580	Safety Evaluation Report on the Westinghouse AP1000 Design Certification Amendment Related to Design Control Document Section 6.1.2.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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8/6/2010	ML102240271	Transmittal of Technical Report APP-GW-GLR-096 Revision 1, and APP-GW-GLR-097 Revision 1, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses."	Sisk, R. B.	Westinghouse	AP1000 APP-GW-GLR-097, Rev 1 AW-10-2901 DCP_NRC_002998
8/11/2010	ML102290202	Summary of the April 19 - 21, 2010, Audit of AP1000 Design Certification - Regulatory Audit Of Open Items: J-Groove Welds and Containment Recirculation Screens.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
8/12/2010	ML102110322	Westinghouse - Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002942
8/13/2010	ML102170183	Audit Plan For Review Of The AP1000 Seismic Margins Issues - August 9 - 10, 2010.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
8/13/2010	ML102280266	Response to Notice of Violation Cited in NRC Inspection Report No. 05200006-2010-202 and (Notice of Violation) Date July 16, 2010.	Sisk, R. B.	Westinghouse	DCP_NRC_003009 IR-10-202
8/16/2010	ML102310235	06/15/2010 - 06/16/2010 Meeting With Westinghouse To Discuss AP1000 Instrumentation And Control System Open Issues To Complete Final Safety Evaluation Report.	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	

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8/16/2010	ML102170455	Input for Safety Evaluation Report Resolving Revision 17 Issues for Section 5.2.3 (Reactor Coolant Pressure Boundary Materials) for AP1000 Design Certification Document.	Terao, D. A.	NRC/NRO/DE/CIB1	
8/16/2010	ML102180160	Westinghouse - Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Sisk, R. B.	Westinghouse	DCP_NRC_002941
8/17/2010	ML102180276	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of The Final Information On Proposed Changes For The AP1000 Design Control Documents Rev.18 (DCP_NRC_002850).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002850
8/17/2010	ML102210275	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of Information On Proposed Changes For The AP1000 Design Control Documents Revision 18.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002918

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8/17/2010	ML102320580	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of Information On Proposed Changes For The AP1000 Design Control Documents Revision 18.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002863
8/17/2010	ML102420175	AP1000 Request For Withholding Information From Public Disclosure, Submittal Of Proprietary And Non-Proprietary Versions Of Information On Proposed Changes For The AP1000 Design Control Documents Revision 18 (DCP_NRC_002925).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002925
8/17/2010	ML102420178	AP1000 Standard COL Technical Report Submittal of APP-GW-GLR-026, Revision 5 (TR44).	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_003010 TR44
8/17/2010	ML102300228	2010/08/17 AP-1000 DCD Review - RE: Seismic Margins OI		- No Known Affiliation	
8/17/2010	ML102300492	2010/08/17 AP-1000 DCD Review - Emailing: OI-SRP9.1.3-SBPA-13 R3.doc		- No Known Affiliation	
8/18/2010	ML102310077	Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, September 20-21, 2010.	Wang, W.	NRC/ACRS	
8/18/2010	ML102420170	09/08/2010 Notice of Planning Meeting With Westinghouse On AP1000 Core Reference Report.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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8/23/2010	ML102250118	Inspection of AP1000 Pressurize Water Reactor Design Aircraft Impact Assessment.	Peralta, J. D.	NRC/NRO/DCIP/CQVP	
8/23/2010	ML102250159	2010/08/23 AP-1000 DCD Review - RE: Fuel rack RAIs Update		- No Known Affiliation	
8/25/2010	ML102250216	Letter - Transmittal of Safety Evaluation for Chapter 5 Titled "Reactor Coolant System and Connected Systems," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/25/2010	ML102250247	SE - Reactor Coolant System and Connected Systems.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/25/2010	ML102390520	Proprietary Information Review - Advanced Final Safety Evaluation for Chapter 13, Titled "Conduct of Operations," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/25/2010	ML102390521	SE - Conduct of Operations.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/25/2010	ML102390522	Submittal of AP1000 Instrumentation and Control Documents to Support of the AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	DCP_NRC_003021
8/25/2010	ML102210227	APP-IIS-J0R-002-NP Revision 0, WCAP-17226-NP R2, "Assessment of Potential Interactions Between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in AP1000 In-Core Instrumentation System."		Westinghouse	APP-IIS-J0R-002-NP, Rev 0 DCP_NRC_003021 WCAP-17226-NP, R2

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8/25/2010	ML102210237	APP-IIS-J0R-001-P, Revision 1, WCAP-17226-P, R2, "Assessment of Potential Interactions Between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000 In-Core Instrumentation System."		Westinghouse	APP-IIS-J0R-002-P, Rev 0 DCP_NRC_003021 WCAP-17226-P, R2
8/26/2010	ML102210335	Letter - Transmittal of Advanced Final Safety Evaluation for Chapter 7 Titled "Instrumentation and Controls," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/26/2010	ML102420177	Enclosure 1 - 7.0 Instrumentation and Control Safety Evaluation.		NRC/NRO	
8/26/2010	ML102290132	Enclosure 2 AP1000 Instrumentation and Control System Secure Development and Operational Environment Safety Evaluation Report.		NRC/NRO	
8/26/2010	ML102420179	2010/08/26 AP-1000 DCD Review - Revised drafts for review: RAI-SRP5.2.3-CIB1-01 R4A and RAI-SRP5.4.1-CIB1-01 R3A		- No Known Affiliation	
8/27/2010	ML102530306	AP1000 Subcommittee Review of Selected Chapters of the Advanced Safety Evaluation Report - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	AP1000
8/30/2010	ML102280208	2010/08/30 AP-1000 DCD Review - AP1000 DCD Capture - 8/30/10		NRC	

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8/30/2010	ML102430008	Representative Draft Slides for September 8, 2010 AP1000 Core Reference Report Planning Meeting with NRC (Proprietary).	Gresham, J. A.	Westinghouse	LTR-NRC-10-58
8/31/2010	ML102430008	Proprietary Information Review - Advanced Final Safety Evaluation For Chapter 18 Titled "Human Factors Engineering," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
8/31/2010	ML102300415	Westinghouse Response to NRC Inspection Report No. 05200006-10-202 and Notice of Violation.	Peralta, J. D.	NRC/NRO/DCIP/CQVP	IR-10-202
8/31/2010	ML102440055	Westinghouse Response to NRC Inspection Report No. 05200006-10-202 and Notice of Violation.	Peralta, J. D.	NRC/NRO/DCIP/CQVP	IR-10-202
9/1/2010	ML102500223	Progress Report on the Review of the AP1000 Design Certification Application.	Matthews, D. B.	NRC/NRO/DNRL	
9/1/2010	ML102500224	Summary of the Audit Conducted on AP1000 Document WCAP-17280-P, at Westinghouse, Rockville, Maryland, on June 28, 2010.	Davis, R. H.	NRC/NRO/DE/CIB2	WCAP-17280-P
9/2/2010	ML102500225	Submittal of AP1000 RCP Casing: Cast Stainless Steel Weld Inspectability 8/26/10 Presentation.	Sisk, R. B.	Westinghouse	AP1000 AW-10-2910 DCP_NRC_003032
9/3/2010	ML102510192	Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003014

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9/3/2010	ML102530307	Letter re: Supplemental Information to Resolve Shield Building Audit Questions.	Ziesing, R. F.	Westinghouse	DCP_NRC_003029
9/3/2010	ML102580174	Shield Building Benchmarking Analysis - (Proprietary).		Westinghouse	DCP_NRC_003029
9/8/2010	ML102310320	AP1000 Core Reference Report Planning Meeting.		Westinghouse	LTR-NRC-10-58
9/10/2010	ML102630020	Westinghouse - Response to Request for Additional Information on SRP Section TR03 in Support of AP1000 Design Certification Amendment Application.	Sisk, R. B.	Westinghouse	AP1000 DCP_NRC_003042 SRP TR03
9/13/2010	ML102570087	Advanced Final Safety Evaluation For Chapter 15 titled: "Transient And Accident Analyses," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
9/13/2010	ML102630017	Westinghouse, AP1000 Response to Request for Additional Information (TR44).	Ziesing, R. F.	Westinghouse	DCP_NRC_003044 TR44
9/14/2010	ML102630018	Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, October 5, 2010.	Wang, W.	NRC/ACRS	
9/15/2010	ML102670213	AP1000 Response to Request for Additional Information (SRP 19).	Ziesing, R. F.	Westinghouse	AP1000 DCP_NRC_003045 SRP 19
9/15/2010	ML102670214	AP1000 Technical Report Review, Enclosure 1 to DCP_NRC_003045.		Westinghouse	AP1000 DCP_NRC_003045 SRP 19
9/15/2010	ML102410009	Westinghouse Submittal of Slides for the AP100 Core Reference Report Planning Meeting on September 8, 2010.	Gresham, J. A.	Westinghouse	LTR-NRC-10-54

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9/15/2010	ML102410012	Slides for the Open Portion of the AP1000 Core Reference Report Planning Meeting on September 8, 2010.		Westinghouse	LTR-NRC-10-54
9/16/2010	ML102630015	Transmittal of Safety Evaluation Report for Chapter 6, "Engineered Safety Features," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
9/16/2010	ML102670216	SE - 6.0 Engineered Safety Features.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
9/16/2010	ML102580168	Westinghouse Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_0030336
9/16/2010	ML102650098	Slides for the Closed Portion of the AP1000 Core Reference Report Planning Meeting on September 8, 2010.		Westinghouse	LTR-NRC-10-54
9/20/2010	ML102650099	Advisory Committee on Reactor Safeguards AP1000 Subcommittee Review of Selected Chapters of the Safety Evaluation Report with Open Items - AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL	
9/20/2010	ML102220446	Transmittal of WEC Shield Building Action Item 21.	Ziesing, R. F.	Westinghouse	AW-10-2909 DCP_NRC_003030
9/22/2010	ML102670162	07/29/10 Summary of Meeting Held With Westinghouse Regarding the Proposed AP1000 Design Certification Amendment, Held in Rockville, Maryland on July 29, 2010.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	

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9/22/2010	ML102510260	Slides and Viewgraphs - Summary Of A Category 1 Public Meeting Held With Westinghouse Regarding The Proposed AP1000 Design Certification Amendment, Held In Rockville, Maryland On July 29, 2010		Westinghouse	
9/22/2010	ML102510281	Westinghouse Electirc Co., Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003035
9/23/2010	ML102650183	08/16/2010 - Audit Report for Review of the AP1000 Hatch Hoist Mount	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
9/23/2010	ML102640391	07/16/2010 - Audit Report for Review of the AP1000 Spent Fuel Pool Heat Load Calculation.	Buckberg, P.	NRC/NRO/DNRL/ NWE2	
9/23/2010	ML102630004	10/7, 10/14, 10/21, and 10/28/2010 Notice of Meeting With Westinghouse on AP1000 Design Certification Amendment - Nuclear Regulatory Commission Meeting on Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	

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9/24/2010	ML102700378	AP1000 Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of AP1000 Responses to Request for Additional Information (RAI-SRP6.2.2-CIB1-31, Revision 1 and RAI-SRP6.2.2-SPCV-25, Revision 2).	Clark, P.	NRC/NRO/DNRL/ NWE2	
9/28/2010	ML102500412	Structural Engineering Branch 1 Input to Safety Evaluation Report for AP1000 Design Control Document, Revision 17, Tier 2, Section 9.1.1.2.1 and 9.1.2.2.1 Application (Phase 4).	Thomas, B. E.	NRC/NRO/DE/SEB1	
9/28/2010	ML102500415	Safety Evaluation for AP1000 Design Control Document, Revision 17, as Amended by APP-GW-GLR-096, Revision 1, Certification Phase 4.	McKirgan, J. B.	NRC/NRO/DSRA/ SPCV	APP-GW-GLR-096, Rev 1
9/29/2010	ML102640352	08/26/2010 Summary of a Category 1 Public Meeting Held with Westinghouse Regarding the AP1000 Reactor Coolant Pump Casing Weld, Held in Rockville, Maryland.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
9/29/2010	ML102660263	08/26/2010 Westinghouse Nonproprietary Presentation "AP1000 RCP Casing: Cast Stainless Steel Weld Inspectability."		Westinghouse	

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9/29/2010	ML102660378	AP1000 Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of WCAP-16914, Revision 5 and WCAP-17028, Revision 6 (DCP_NRC_002931).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002931 WCAP-16914, Rev 5 WCAP-17028, Rev 6
9/29/2010	ML102670260	AP1000, Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of Supplemental Information to Resolve Shield Building Audit Questions (DCP_NRC_003029).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003029
9/29/2010	ML102770328	AP1000, Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of AP1000 RCP Casing: Cast Stainless Steel Well Inspectability 08/26/2010 Presentation (DCP_NRC_003032).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003032
9/29/2010	ML102770447	AP1000, Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of "AP1000 Shield Building - Benchmarking, Analysis, Testing and Design Overview," (DCP_NRC_002995).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002995

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9/29/2010	ML102780269	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-005, Revision 4 (TR-09).	Ziesing, R. F.	Westinghouse	DCP_NRC_003051 TR-09
9/29/2010	ML102740287	Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003015
9/29/2010	ML102740297	Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003048
9/30/2010	ML102780270	Chapter 9 - Auxiliary Systems.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
9/30/2010	ML102790599	Letter - Proprietary Information Review - Advanced Final Safety Evaluation For Chapter 9 Titled "Auxiliary Systems," Of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	Buckberg, P. H. McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
9/30/2010	ML102790613	Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003034
9/30/2010	ML102800192	Submittal of APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building".	Ziesing, R. F.	Westinghouse	APP-1200-S3R-003, Rev 3 AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800199	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building".		Westinghouse	AW-10-2958 DCP_NRC_002053
9/30/2010	ML102800217	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Cover to Section 6.		Westinghouse	AW-10-2958 DCP_NRC_003053

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9/30/2010	ML102800222	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Section 7: Shield Building Test Program.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800225	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Section 7.9: Pushout Test through Section 7.14: References.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800227	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Section 8: Benchmarking to Section 9.11: References.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800230	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Section 10: Shield Building Design Process to Appendix A.4: References.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800237	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Appendix B through Figures.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800240	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Appendix C: Level 3 Analyses.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102800243	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Appendix C.5.5: Roof Rebars.		Westinghouse	AW-10-2958 DCP_NRC_003053
9/30/2010	ML102740252	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Appendix D: Design Sketches.		Westinghouse	AW-10-2958 DCP_NRC_003053

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9/30/2010	ML102740296	APP-1200-S3R-003, Rev. 3, "Design Report for the AP1000 Enhanced Shield Building," Appendix H.2 to End.		Westinghouse	AW-10-2958 DCP_NRC_003053
10/1/2010	ML102660392	Transmittal of Safety Evaluation for AP1000 Design Change Package 73.	McKirgan, J. B.	NRC/NRO/DSRA/ SPCV	
10/1/2010	ML102660432	Transmittal of Safety Evaluation for AP1000 Design Change Package 75.	McKirgan, J. B.	NRC/NRO/DSRA	
10/4/2010	ML102740302	AP1000 Request for Withholding Additional Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of "Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18" (DCP_NRC_002874).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002874
10/4/2010	ML102770180	AP1000 Request for Withholding Information from Public Disclosure, Submittal of Proprietary and Non-Proprietary Versions of "Final Information on Proposed Changes for the AP1000 DCD Rev. 18 Update Summary of New Change Notices" (DCP_NRC_002884).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002884
10/4/2010	ML102770184	AP1000 Subcommittee Review Of Selected Chapters Of The Advanced Safety Evaluation Report- AP1000 Design Certification Amendment.	Mathews, D. B.	NRC/NRO/DNRL	

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10/4/2010	ML102770643	Memo: Structural Engineering Branch 1 Input to Safety Evaluation Report for AP1000 Design Control Document, Revision 17 (Phase 4).	Thomas, B. E.	NRC/NRO/DE/SEB1	
10/4/2010	ML102810225	Enclosure: AP1000 Design Control Document, Revision 17, Safety Evaluation Report for Standard Review Plan Sections 3.7.1, 3.7.2, and 3.7.3.	Thomas, B. E.	NRC/NRO/DE/SEB1	
10/4/2010	ML102810226	Safety Evaluation Report Input on Proposed Revision 18 Changes to AP1000 Design Control Document Related to American Society of Mechanical Engineers Code Edition and Addenda from Tier 2* to Tier 2.	Terao, D. A.	NRC/NRO/DE/CIB1	
10/5/2010	ML103010111	Informs That Westinghouse Has Completed Proprietary Review of Advanced Final Safety Evaluation for Chapter 7, "Instrumentation & Controls," in NUREG-1793.	Ziesing, R. F.	Westinghouse	AW-10-2959 DCP_NRC_003054 NUREG-1793
10/5/2010	ML102730729	Westinghouse Markup of NRC Chapter 7 AFSE.		Westinghouse	AW-10-2959 DCP_NRC_003054 NUREG-1793
10/5/2010	ML102800603	Transcript of the Advisory Committee on Reactor Safeguards Subcommittee on AP1000 - October 5, 2010 (Open Session)	Wang, W.	NRC/ACRS	NRC-471
10/6/2010	ML102870236	Meeting Summary for the Planning Meeting with Westinghouse on AP1000 Core Reference Report.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	

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10/13/2010	ML103430502	Chapter 19 - Severe Accidents.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
10/13/2010	ML102310352	Proprietary Information Review - Advanced Final Safety Evaluation For Chapter 19 Titled "Severe Accidents," Of NUREG-1793, Supplement 2 - Ap1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
10/13/2010	ML103120263	Chapter 3 - Design of Structures, Components, Equipment, and Systems.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
10/14/2010	ML103120264	Chapter 15 - Transient and Accident Analyses.		NRC/NRO/DNRL/ NWE2	NUREG-1793
10/14/2010	ML103120265	Westinghouse - Chapter 15 "Transient and Accident Analyses" SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003061
10/14/2010	ML102870071	Westinghouse, Letter Regarding Chapter 14 SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003058
10/14/2010	ML102920143	Westinghouse - Chapter 13 "Conduct of Operation" SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003060
10/15/2010	ML102920343	AP1000 Subcommittee Review of Selected Chapters of The Advanced Safety Evaluation Report- AP1000 Design Certification Amendment.	Matthews, D. B.	NRC/NRO/DNRL/ NWE2	
10/15/2010	ML102910231	Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003050
10/15/2010	ML102930086	Westinghouse - Chapter 5 "Reactor Coolant System and Connected Systems," SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003059

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10/18/2010	ML102930087	2010/10/18 AP-1000 DCD Review - RE: Error occurred in Process Hearings		NRC/OIS	
10/18/2010	ML102930088	Submittal of Supplementary Information on Proposed Changes for the AP1000 Design Control Document, Rev. 18.	Ziesing, R. F.	Westinghouse	AW-10-2962 DCP_NRC_003066
10/18/2010	ML102940228	G20100648/LTR-10-0465/EDATS: SECY-2010-0498 - Ltr. Louis Zeller Re: Request to Overturn Withholding of Information from Public Disclosure and a Call for a Special Investigation.	Zeller, L. A.	Blue Ridge Environmental Defense League	G20100648 LTR-10-0465 SECY-2010-0498
10/19/2010	ML102940229	AFSE - 23 - Design Changes Proposed in Accordance with ISG-11.		NRC/NRO	
10/19/2010	ML102930106	Enclosure 3 - Westinghouse Markup of NRC Chapter 6 ASE Review Indicating Proprietary Sections.		Westinghouse	DCP_NRC_003067
10/19/2010	ML102950077	Westinghouse - Chapter 6 ASE Review.	Ziesing, R. F.	Westinghouse	AW-10-2986 DCP_NRC_003067
10/20/2010	ML102920565	Minutes of the Advisory Committee on Reactor Safeguards Subcommittee on the AP1000 Reactor, July 21-22, 2010, Rockville, Maryland.	Wang, W.	NRC/ACRS	
10/20/2010	ML103010046	Westinghouse, Supplementary Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003068

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10/21/2010	ML103010289	Letter - Proprietary Information Review - Advanced Final Safety Evaluation for Chapter 23, Titled "Design Changes Proposed in Accordance with ISG-11," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
10/22/2010	ML103010295	Westinghouse, Response to Request for Additional Information (TR85) in support of the AP1000 Design Certification Amendment Application.	Ziesing, R. F.	Westinghouse	
10/26/2010	ML103010450	Westinghouse, Regarding Chapter 16 SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003077
10/26/2010	ML103010477	Letter from R. F. Ziesing re: Westinghouse regarding the ommission of proprietary data in the Safety Evaluation Report.	Ziesing, R. F.	Westinghouse	DCP_NRC-003073
10/26/2010	ML103010481	Westinghouse, Review of SAMDA Relative to Impact for AP1000 DCD Revision 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003057
10/26/2010	ML103010484	Westinghouse, Letter Regarding Chapter 4 SER Review and Non Proprietary Determination.	Ziesing, R. F.	Westinghouse	DCP_NRC_003074
10/26/2010	ML102710493	Westinghouse, Chapter 12 SER Review and Non-Proprietary Determination Letter.	Ziesing, R. F.	Westinghouse	DCP_NRC_003076
10/26/2010	ML103000063	Westinghouse, Chapter 11 SER Review and Non-Proprietary Determination Letter.	Ziesing, R. F.	Westinghouse	DCP_NRC_003075

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10/27/2010	ML103000144	Advanced Final Safety Evaluation for Chapter 3, Titled "Design of Structures, Components, Equipment, and Systems," of NUREG-1793, Supplement 2-AP1000 Design Certification Amendment.	Gleaves, B. C.	NRC/NRO/DNRL/ NWE2	
10/27/2010	ML101241063	Federal Register Notice Regarding the ACRS Subcommittee Meeting on AP1000, December 1, 2010.	Wang, W.	NRC/ACRS	
10/27/2010	ML102850096	11/18/2010 Notice of Closed Meeting with Westinghouse to Discuss Their October 7, 2010 Proprietary Response to NRC Regulatory Issue Summary 2007-08 Regarding the AP1000 Updated Supplier Listing.	Kavanagh, K. A.	NRC/NRO/DCIP/CQVP	
10/28/2010	ML102980583	P4 Chapter 3 SE Not Including 3.7 and 3.8.		NRC/NRO/DNRL/ NWE2	
10/28/2010	ML103060042	Memo - AP1000 Design Certification Amendment - Advanced Safety Evaluation, Chapter 3, "Design of Structures, Components, Equipment, and Systems".	Matthews, D. B.	NRC/NRO/DNRL	
10/28/2010	ML103020551	IR 05200006-10-203, and Notice of Violation on 09/27/2010 - 10/01/2010, Westinghouse AP1000 Pressurized Water Reactor Design Aircraft Impact Assessment Inspection.	Rasmussen, R. A.	NRC/NRO/DCIP/CQVP	
10/28/2010	ML103090320	Westinghouse - Chapter 19 SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003080

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10/29/2010	ML102950122	10/29/10 Acknowledgement Ltr to L. Zeller re Request to Release Information Regarding Westinghouse's AP1000 Nuclear Power Reactor.	Vietti-Cook, A. L.	NRC/SECY	
11/1/2010	ML103410114	2010/11/01 AP-1000 DCD Review - FW: NRC Concern with AO/111% Overspeed Trip		- No Known Affiliation	
11/2/2010	ML103120399	Letter - Request for Withholding Information from Public Disclosure, Submittal of Proprietary Version of "AP1000 Plant Startup Human Factors Engineering Design Verification Plan" (DCP_NRC_002990).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002990
11/2/2010	ML103120400	Transcript of the Advisory Committee on Reactor Safeguards Westinghouse AP1000 DCD Subcommittee (OPEN Session) on November 2, 2010 in Rockville, MD. Pages 1-73.		NRC/ACRS	NRC-529
11/4/2010	ML103090319	Westinghouse - Chapter 23 AFSE Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003081
11/4/2010	ML103090321	Westinghouse Markup of NRC Chapter 23 AFSE Replacing Proprietary Information with Non-Proprietary Information, Enclosure 3.		Westinghouse	DCP_NRC_003081
11/5/2010	ML103350684	2010/11/05 AP-1000 DCD Review - RE: NRC Concern with AO/111% Overspeed Trip		- No Known Affiliation	
11/5/2010	ML102870605	2010/11/05 AP-1000 DCD Review - AP1000 DCD Capture - 11/5/10		NRC/NRO	

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11/5/2010	ML103060229	Transcript of ACRS 567th Meeting, November 5, 2009, Pages 1-250.		NRC/ACRS	NRC-3181
11/8/2010	ML103050310	P4 Chapter 3, Sections 3.7 and 3.8 With Shield Building (Section 3.8.4.1.1).		NRC/NRO	
11/8/2010	ML103190365	Proprietary Information Review - Advanced Final Safety Evaluation For Sections 3.7 and 3.8 Titled, "Seismic Design," and "Design Of Category I Structures," Respectively, of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793 S2
11/9/2010	ML103190367	Memo to ACRS Transmitting Sections Sections 3.7 & 3.8, with 3.8.4 of the AP1000 Design Certification Amendment - Advanced Final Safety Evaluation, "Design Of Structures, Components, Equipment, And Systems".	Matthews, D. B.	NRC/NRO/DNRL	NUREG-1793, Suppl 2
11/9/2010	ML103190453	Westinghouse - Chapter 18 of Advanced Final Safety Evaluation (AFSE) Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003055
11/9/2010	ML103190454	Chapter 18 of Advanced Final Safety Evaluation (AFSE) Review.		Westinghouse	DCP_NRC_003055
11/9/2010	ML103000111	Westinghouse, Determination of Chapter 8 SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003085
11/9/2010	ML103210409	Applicant's Supplemental Environmental Report - Amendment to Standard Design Certification.	Ziesing, R. F.	Westinghouse	DCP_NRC_003086

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11/12/2010	ML103210455	G20100648/LTR-10-0465/EDATS: SECY-2010-0498 - Louis A. Zeller Ltr. Re: Request to Overturn Withholding of Information from Public Disclosure and a Call for a Special Investigation.	Johnson, M. R.	NRC/NRO	G20100648 LTR-10-0465 SECY-2010-0498
11/12/2010	ML102980545	Reply to Notice of Violation Cited in NRC Inspection Report No. 05200006/2010-203 dated October 28, 2010.	Ziesing, R. F.	Westinghouse	DCP_NRC_003084 IR 10-203
11/12/2010	ML102980562	Westinghouse - Response to Request for Additional Information on SRP Section 10 in Support of AP1000 Design Certification Amendment Application.	Ziesing, R. F.	Westinghouse	DCP_NRC_003088 SRP 10
11/17/2010	ML103000381	Transmittal of Advanced Safety Evaluation for Chapter 2 "Site Envelope" of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/NWE2	NUREG-1793 S2
11/17/2010	ML103230126	Chapter 2 Advanced Safety Evaluation Titled "Site Envelope".	McKenna, E. M.	NRC/NRO/DNRL/NWE2	NUREG-1793 S2
11/17/2010	ML103230129	Chapter 2 Advanced Safety Evaluation Titled "Site Envelope" Phase 4 SE Concurrence Sheet.		NRC/NRO	
11/17/2010	ML103230439	Westinghouse, Response for Additional Information Related to an AP1000 Design Enhancement that Mitigates the Risk of Spurious Actuation of Automatic Depressurization System.	Ziesing, R. F.	Westinghouse	DCP_NRC_003090

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11/17/2010	ML103230472	Westinghouse - Final Information on Proposed Changes for the AP1000 Design Control Document Rev. 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003087
11/17/2010	ML103230507	Advanced Final SER Section 307 - Seismic Design.		NRC/ACRS	
11/17/2010	ML103260235	Advanced Final SER Section 3.8 - Design of Category 1 Structures; Westinghouse AP1000 Design Certification Amendment Applicaiton Review.		NRC/NRO	
11/17/2010	ML103260236	Advanced Final SE Chapter 3.8.4 Other Category I Structures; Westinghouse AP1000 Design Certification Amendment Application Review.		NRC/NRO	
11/17/2010	ML103470497	Westinghouse - Chapter 23 ASE Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003089
11/17/2010	ML102650378	Westinghouse Markup of NRC Chapter 23 AFSE Replacing Proprietary Information with Non-Proprietary Information. (Proprietary).		Westinghouse	AW-10-2989 DCP_NRC_003089 NUREG-1793
11/17/2010	ML102650626	Transcript of the Advisory Committee on Reactor Safeguards AP1000 Subcommittee - November 17, 2010 (OPEN). Pages 1-110.	Wang, W.	NRC/ACRS	NRC-558
11/18/2010	ML102650629	07/23/2010 Summary of a Public Meeting with Westinghouse Regarding Change Notice 74 and the AP1000 Containment Vessel External Pressure Design	Donnelly, P. M.	NRC/NRO/DNRL/ NWE2	

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11/18/2010	ML103120238	AP1000 Containment Vessel External Pressure Analysis And Design.		NRC/NRO	
11/18/2010	ML103120290	AP1000 Containment Vessel External Pressure Analysis And Design.		Westinghouse	
11/18/2010	ML103120348	AP1000 Request for Withholding Information from Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of Westinghouse Shield Building Action Item 21 (DCP_NRC_003030).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
11/18/2010	ML103200351	Letter - AP1000 Request for Withholding Information from Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of Shield Building Benchmarking Analysis (DCP_NRC_003029).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
11/18/2010	ML103230409	Letter - AP1000 Request for Withholding Information from Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of Design Report for the AP1000 Enhanced Shield Building (DCP_NRC_003053).	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
11/18/2010	ML103230422	Transmittal of Safety Evaluation for AP1000 Design Change Package 64.	McKirgan, J. B.	NRC/NRO/DSRA/ SPCV	
11/18/2010	ML103230424	11/18/2010 QA Meeting Agenda.	Kavanagh, K. A.	NRC/NRO/DCIP/CQVP	
11/18/2010	ML103340044	11/18/2010 AP1000 EQ Program	Kavanagh, K. A.	NRC/NRO/DCIP/CQVP	
11/18/2010	ML103340047	11/18/2010 EDV Discussion.		NRC/NRO/DCIP/CQVP Westinghouse	

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11/18/2010	ML103230437	GSI-191 ACRS Slide Presentation - AP1000 Long-Term Cooling Debris Issue Resolution. (Non-Proprietary Version).	Schulz, T. L.	Westinghouse	DCP_NRC_003096 GSI- 191
11/18/2010	ML103340045	GSI-191 ACRS Slide Presentation - AP1000 Long-Term Cooling Debris Issue Resolution. (Proprietary Version).	Schulz, T. L.	Westinghouse	DCP_NRC_003096 GSI-191
11/19/2010	ML103340048	Minutes of the ACRS Subcommittee on the AP1000 Reactor. September 20-21, 2010, Rockville, Maryland.	Wang, W.	NRC/ACRS	
11/19/2010	ML103470566	GSI-191 ACRS Slide Presentation - AP1000 Long-Term Cooling Debris Issues Resolution Supplemental Information. (Non-Proprietary Version).	Schulz, T. L.	Westinghouse	DCP_NRC_003096 GSI-191
11/19/2010	ML103470567	GSI-191 ACRS Slide Presentation - AP1000 Long-Term Cooling Debris Issues Resolution Supplemental Information. (Proprietary Version).	Schulz, T. L.	Westinghouse	DCP_NRC_003096 GSI-191
11/19/2010	ML103330200	AP1000 Response to Request for Open Item (SRP 19), RAI-SRP19F-AIA-01 R3 and RAI-SRP19F-AIA-09 R2 In Support of the Design Certification Amendment Application.	Ziesing, R. F.	Westinghouse	DCP_NRC_003093
11/21/2010	ML103200633	Westinghouse - Shield Building Figures and PRA Discussion.	Ziesing, R. F.	Westinghouse	DCP_NRC_003082
11/21/2010	ML103300210	Westinghouse - Shield Building Figures.		Westinghouse	DCP_NRC_003082

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11/22/2010	ML103300211	Transmittal of Safety Evaluation for AP1000 Design Change Package 74.	McKirgan, J. B.	NRC/NRO/DSRA/ SPCV	
11/22/2010	ML103330185	AP1000 Response to Request for Additional Information (SRP 19).	Ziesing, R. F.	Westinghouse	DCP_NRC_003094 RAI-SRP19F-AIA-01, Rev 4
11/22/2010	ML103330229	Westinghouse - Re-submittal of Chapter 6 AFSER Proprietary Review.	Ziesing, R. F.	Westinghouse	AW-10-3026 DCP_NRC_003095
11/22/2010	ML103340060	Enclosure 3 - Westinghouse Markup of NRC Chapter 6 ASE Review Indicating Proprietary Sections.		Westinghouse	DCP_NRC_003095
11/22/2010	ML103340065	Submittal of Proprietary and Non-Proprietary Version of WCAP-16675, Revision 4, "AP1000 Protection & Safety Monitoring System Architecture Technical Report."	Ziesing, R. F.	Westinghouse	AW-10-2988 DCP_NRC_003091 WCAP-16675, Rev 4
11/22/2010	ML103070212	Westinghouse, Chapter 3 AFSE Proprietary Information Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003083
11/22/2010	ML103080068	Enclosure 3, Westinghouse Markup of NRC Chapter 3 (Section 3.7 & 3.8) AFSER Indicating Proprietary Sections.		Westinghouse	DCP_NRC_003083
11/23/2010	ML103260447	Transmittal of Safety Evaluation Report for Chapter 1, Titled "Introduction and General Discussion," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
11/23/2010	ML103340046	Chapter 1, "Introduction and General Discussion."	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2

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11/23/2010	ML103340049	Westinghouse Response To U.S. Nuclear Regulatory Commission (NRC) Inspection Report [05200006/2010-203] and Notice of Violation.	Rasmussen, R. A.	NRC/NRO/DCIP/CQVB	
11/23/2010	ML103370228	Submittal of GSI-191 ACRS Slide Presentation - November 18-19, 2010 (Proprietary).	Ziesing, R. F.	Westinghouse	DCP_NRC_003096
11/23/2010	ML103370230	Report of Audits for AP1000 Design Change Package Nos. 72 and 74.	Mrowca, L. A.	NRC/NRO/DSRA/ SPLA	
11/29/2010	ML103370231	AP1000 Shield Building - Advisory Committee on Reactor Safeguards -November 17, 2010 - (Non-Proprietary).		Westinghouse	DCP_NRC_003097
11/29/2010	ML103370232	Submittal of ACRS November 17 and 18, 2010 Presentation Materials.	Ziesing, R. F.	Westinghouse	DCP_NRC_003097
11/29/2010	ML103370233	AP1000 Design Control Document Amended Design - Section 3.7 - Seismic Design Presentation (November 17, 2010) - (Non-Proprietary).		Westinghouse	DCP_NRC_003097
11/29/2010	ML103370234	AP1000 Design Control Document Amended Design-Section 3.8 - Design of Category I Structures Presentation (November 17, 2010) - Non-Proprietary).		Westinghouse	DCP_NRC_003097
11/29/2010	ML103370195	AP1000 Shield Building - Advisory Committee on Reactor Safeguards - November 17, 2010 - (Proprietary).		Westinghouse	DCP_NRC_003097
11/29/2010	ML103370196	Shield Building Additional Comments (November 18, 2010) - (Proprietary).		Westinghouse	DCP_NRC_003097

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11/30/2010	ML103370194	WCAP-16675-NP, Rev. 5, App-GW-GLR-147, Rev. 2, "AP1000 Protection and Safety Monitoring System Architecture Technical Report."		Westinghouse	DCP_NRC_003099
11/30/2010	ML103370220	WCAP-16675-P, Rev. 5, App-GW-GLR-071, Rev. 5, "AP1000 Protection and Safety Monitoring System Architecture Technical Report."	Schindhelm, E. P. Single, R. E.	Westinghouse	DCP_NRC_003099
12/1/2010	ML103370587	Submittal of Proprietary and Non-Proprietary Versions of WCAP-16592, Revision 2.	Ziesing, R. F.	Westinghouse	DCP_NRC_003100 WCAP-16592, Rev 2
12/1/2010	ML103480059	Submittal of Proprietary and Non-Proprietary Versions of WCAP-16675, Revision 5.	Ziesing, R. F.	Westinghouse	AW-10-3029 DCP_NRC_003099 WCAP-16675, Rev 5
12/1/2010	ML103480123	Westinghouse Chapter 2 SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003078
12/1/2010	ML103480124	Westinghouse Updated Application to Amend the AP1000 Design Certification Rule.	Rupprecht, S. D.	Westinghouse	DCPNRC_003098
12/1/2010	ML103480125	Westinghouse AP1000 Cover Letter Rev. 18 - Packing Slip	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.CVR.S APP-GW-GL-700.CVR.S.18 APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSIO N.8
12/1/2010	ML103480127	Westinghouse AP1000 Cover Letter Rev. 18 - Cover and Copyright	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.CVR.S APP-GW-GL-700.CVR.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSIO N.8

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12/1/2010	ML103480128	Westinghouse AP1000 Cover Letter Rev. 18 - Revision 18 Transmittal Letter	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.CVR.S APP-GW-GL-700.CVR.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8
12/1/2010	ML103480130	Westinghouse AP1000 Design Control Document Rev. 18 - Introduction	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8
12/1/2010	ML103480149	Westinghouse AP1000 Design Control Document Rev. 18 - Tier 1 - Change Roadmap	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8
12/1/2010	ML103480150	Westinghouse AP1000 Design Control Document Rev. 18 - Tier 1 - List of Effective Pages	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8
12/1/2010	ML103480151	Westinghouse AP1000 Design Control Document Rev. 18 - Tier 2 - List of Effective Pages	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8
12/1/2010	ML103370648	Westinghouse AP1000 Design Control Document Rev. 18 - Tier 2 - Change Pages	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8
12/1/2010	ML102880708	Westinghouse AP1000 Design Control Document Rev. 18 - Tier 2 - Master Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.S APP-GW-GL-700.DCD.S.18 WESTINGHOUSE WESTINGHOUSE.SUBMISSION.8

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12/3/2010	ML103430482	Redacted Version of Dissenting View on AP1000 Shield Building Safety Evaluation Report With Respect to the Acceptance of Brittle Structural Module to be Used for the Cylindrical Shield Building Wall.	Bergman, T. A.	NRC/NRO/DE	
12/9/2010	ML103480064	Summary of AP1000 Design Certification-Regulatory On-Site and Off-Site Reviews of Open Items for the Westems Computer Code.	Clark, P. M.	NRC/NRO/DNRL/ NWE2	
12/9/2010	ML103480067	12/09/2010 CIP Category III Public Workshop - Managing ITAAC Surge (WEC).	Bedford, B.	Westinghouse	
12/9/2010	ML103480078	Westinghouse, ADS Spurious Actuation - Revision to Failure Modes and Effects Analysis.	Ziesing, R. F.	Westinghouse	DCP_NRC_003102
12/9/2010	ML103420306	Westinghouse - Chapter 1 SER Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003101
12/9/2010	ML103440283	Westinghouse, Submittal of Pages that Contain the Revised Proprietary Markings and a Table Containing NRC Comments and the Associated Westinghouse Responses.	Ziesing, R. F.	Westinghouse	AW-10-3039 DCP_NRC_003103
12/10/2010	ML103440547	Chapter 6 - Engineered Safety Features.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
12/10/2010	ML103480065	Chapter 23 - Design Changes Proposed in Accordance with ISG-11.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
12/10/2010	ML103480072	Chapter 18 - Human Factors Engineering.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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12/10/2010	ML103480073	Westinghouse request for NRC to retire the AP1000 Final Design Approval (FDA) issued March 10, 2006 upon completion of rulemaking for the amendment to the AP1000 NRC certified design.	Ziesing, R. F.	Westinghouse	DCP_NRC_003079
12/10/2010	ML103350501	Proprietary Review of NRC WESTEMS Computer Code Review Summary.	Ziesing, R. F.	Westinghouse	DCP_NRC_003062
12/10/2010	ML103410351	Proprietary Review of WESTEMS Computer Code Review Summary Report.		Westinghouse	DCP_NRC_003062
12/13/2010		AP1000 Technical Reports (Appendix).		NRC/NRO	
12/13/2010		Report on the Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document.	Abdel-Khalik, S.	NRC/ACRS	
12/14/10	ML103440465	12/30/2010-Notice of Meeting with Westinghouse Regarding AP1000 Design Certification Amendment Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
12/14/10	ML103440485	01/06/2011-Notice of Forthcoming Public Meeting with Westinghouse Regarding AP1000 Design Certification Amendment Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	

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12/14/10	ML103440499	01/13/2011-Notice of Forthcoming Public Meeting with Westinghouse Regarding AP1000 Design Certification Amendment Selected Technical or Regulatory Issues.	Jaffe, D. H.	NRC/NRO/DNRL/ NWE2	
12/15/10	ML103550589	Amended Chapter 7 SER Proprietary Review, No Proprietary Information is Included in the ASER.	Ziesing, R. F.	Westinghouse	DCP_NRC_003106
12/16/10	ML103430022	11/17/2010 Summary of Meeting with the Industry Focus Group on New Reactor Operator Licensing.	Pelton, R. M.	NRC/NRO/DCIP/COLP	+reviewedgfw
12/16/10	ML103510336	Non-Concurrence - Advanced Final Safety Evaluation Report for the AP1000 Standard Design Certification Amendment - Chapter 7, "Instrumentation and Control." Non-Concurrence - Advanced Final Safety Evaluation Report for the AP1000 Standard Design Certification	Jackson, T. Roggenbrodt, W. A.	NRC/NRO/DE	
12/16/10	ML103550567	Westinghouse, Change Matrix to review "AP1000 Design Certification Document, Revision 18" (DCD R18).	Ziesing, R. F.	Westinghouse	DCP_NRC_003105
12/17/10	ML103620334	Non-Concurrence - Dissenting View on the AP1000 Instrumentation and controls Designs Certification Amendment Advanced Final Safety Evaluation Report.	Mott, K. D.	NRC/NRO/DE/ICE1	
12/20/10	ML103410348	Long-Term Core Cooling for the Westinghouse AP1000 Pressurized Water Reactor.	Abdel-Khalik, S.	NRC/ACRS	

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12/20/10	ML103570064	Westinghouse, Chapter 7 Non-Concurrence Proprietary Review.	Ziesing, R. F.	Westinghouse	NRC_NRC_003107
12/20/10	ML103570138	G20100750/LTR-10-0537/EDATS: SECY-2010-0619 - Ltr. Said Abdel-Khalik re: Long-Term Core Cooling for the Westinghouse AP1000 Pressurized Water Reactor	Abdel-Khalik, S.	NRC/ACRS	EDATS: SECY-2010-0619 G20100750 LTR-10-0537 SECY-2010-0619
12/21/10	ML103440347	01/10-14-2011, Public Meeting with Westinghouse on AP1000 Design Certification Amendment - Design Control Document Tier 2* Information.	Gleaves, B.	NRC/NRO/DNRL/NWE2	01/14/11
12/28/10	ML103510080	Chapter 7 - Instrumentation and Control.	McKenna, E. M.	NRC/NRO/DNRL/NWE2	
12/28/10	ML103620506	Redacted Version of Non-Concurrence - Insufficient Diversity and Independence in the Implementation Process for AP1000 Instrumentation and Controls Systems and Associated Staff Response.	Jackson, T. Roggenbrodt, W. A.	NRC/NRO/DE	
12/29/10	ML103630486	Redacted Version of the Non-Concurrence Dissenting View on the AP1000 Instrumentation and Controls Design Cert. Amendment Advanced Final Safety Eval. Report with Respect to Eval. and Acceptance Methods Used for Demonstrating Conformance to....	Mott, D. B.	NRC/NRO/DE/ICE1	
01/03/11	ML103000397	SECY-11-0002 - Proposed Rule: AP1000 Design Certification Amendment (RIN 3150-AI81)	Borchardt, R. W.	NRC/EDO	EDATS: NRO-2010-0021 NRO-2010-0021 RIN 3150-AI81

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01/03/11	ML103000412	SECY-11-0002 - Enclosure 1 - Federal Register Notice - Proposed Rule - AP1000 Design Certification Amendment (RIN 3150-AI81)	Vietti-Cook, A. L.	NRC/SECY	EDATS: NRO-2010-0020, RIN 3150-AI81 NRO-2010-0021 SECY-11-0002
01/03/11	ML103000415	SECY-11-0002 - Enclosure 2 - Environmental Assessment - Proposed Rule - AP1000 Design Certification Amendment (RIN 3150-AI81)		NRC/SECY	EDATS: NRO-2010-0021; RIN 3150-AI81 NRO-2010-0021 SECY-11-0002
01/03/11	ML110050388	Markup of Chapter 7 Digital I&C Non-Concurrence.		Westinghouse	
01/03/11	ML110050391	Westinghouse, Chapter 7 Digital I&C Non-Concurrence Proprietary Review.	Ziesing, R. F.	Westinghouse	DCP_NRC_003110
01/05/11	ML110070359	Scaling Calculation for Time to Steady State PCS Film Coverage for the AP1000 Containment Pressure and Temperature Response Analyses.	Ziesing, R. F.	Westinghouse	DCP_NRC_003112
01/19/11	ML110210462	G20110047/LTR-11-0021/EDATS: SECY-2011-0029 - Ltr. Said Abdel-Khalik re: Report on the Safety Aspects of the Aircraft Impact Assessment for the Westinghouse AP1000 Design Certification Amendment Application.	Abdel-Khalik, S. I.	NRC/ACRS	G20110047 LTR-11-0021 SECY-2011-0029
01/26/11	ML110130158	G20100750/LTR-10-0537/EDATS: SECY-2010-0619, Letter to Said Abdel-Khalik, Long-Term Cooling for the Westinghouse AP1000 Pressurized Water Reactor.	Borchardt, R. W.	NRC/EDO	G20100750 LTR-10-0537 SECY-2010-0619

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01/28/11	ML110330046	AP1000 Response to Request for Additional Information (CI-SRP9.1.1 -SRSB-01 and RAI-SRP4.3-SRSB-03).	Ziesing, R. F.	Westinghouse	AW-11-3067 CI-SRP9.1.1-SRSB-01 DCP_NRC_003114 RAI-SRP4.3-SRSB-03
01/31/11	ML110330069	SECY-11-0002 - Chairman Jaczko vote and comments on SECY-11-0002 - Proposed Rule: AP1000 Design Certification Amendment (RIN 3150-AI81)	Jaczko, G. B.	NRC/Chairman	SECY-11-0002
01/31/11	ML110691052	APP-GW-GLR-115 Revision 3, (TR115) "Effect of High Frequency Seismic Content on SSCs".	Lapay, W. S.	Westinghouse	DCP_NRC_003153
02/02/11	ML110350578	CI-23-22, Section 9.2.2.2 and 9.2.2.4.5.2 Wording on CCS Isolation Valve Closure.	Ziesing, R. F.	Westinghouse	DCP_NRC_003119
02/03/11	ML110130185	Ltr. to J. Gresham - AP1000 Request for Withholding Information from Public Disclosure, Transmittal of Proprietary and Non-Proprietary Versions of ACRS November 17 and 18, 2010, Presentation Materials "AP1000 Shield Building Advisory Committee".	Clark, P. M.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003097
02/03/11	ML110390041	Transmittal of Non-Proprietary Document - WCAP-16779-NP, Revision 0, "Overpressure Protection Report for AP1000 Nuclear Power Plant".	Ziesing, R. F.	Westinghouse	DCP_NRC_003120

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02/05/11	ML103560484	G20100734/LTR-10-0528/EDATS: SECY-2010-0595 - Ltr. to Said Abdel-Khalik Re: Report on the Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document.	Borchardt, R. W.	NRC/EDO	G20100734 LTR-10-0528 SECY-2010-0595
02/05/11	ML103560502	Enclosure - Staff Response to Comments in the Advisory Committee on Reactor Safeguards Letter on the AP1000 Design Certification Application.	Abdel-Khalik, S.	NRC/ACRS	G20100734 LTR-10-0528 SECY-2010-0595
02/09/11	ML110400358	SRM-SECY-11-0002 - Proposed Rule: AP1000 Design Certification Amendment (RIN 3150-AI81).	Vietti-Cook, A. L.	NRC/SECY	RIN 3150-AI81 SECY-11-0002
02/09/11	ML110450098	Transmittal of Appendix 19F Security Related Information.	Ziesing, R. F.	Westinghouse	DCP_NRC_003126
02/09/11	ML110450100	Westinghouse - DCD Reference of WCAP-16651-P.	Ziesing, R. F.	Westinghouse	AP1000 DCP_NRC_003123 WCAP-16651-P
02/09/11	ML110450101	AP1000 Response to Request for Additional Information (SRP 3).	Ziesing, R. F.	Westinghouse	AP1000 DCP_NRC_003124 SRP 3
02/09/11	ML110450102	Westinghouse, Response to an NRC Confirmatory item on Chapter 2, CI-SRP2.2-RSAC-01, Section 6.4.4 Wording on Protection of the Operators in the MCR From Offsite Toxic Gas Releases.	Ziesing, R. F.	Westinghouse	DCP_NRC_003118

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02/10/11	ML110190380	Letter - Revision To The Advanced Final Safety Evaluation For Chapter 7 Titled "Instrumentation And Controls," of NUREG-1793, Supplement 2 - AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	NUREG-1793, Suppl 2
02/10/11	ML110190411	Safety Evaluation Report - Section 7.0 Instrumentation And Control.		NRC/NRO/DNRL/ NWE2	
02/10/11	ML110450103	Westinghouse, Tier 2 Section 3.9 Editorial Corrections.	Ziesing, R. F.	Westinghouse	DCP_NRC_003122
02/10/11	ML110450506	Letter from R. F. Ziesing re: Referenced Version of Regulatory Guide 1.82.	Ziesing, R. F.	Westinghouse	DCP_NRC_003125 RG-1.082
02/15/11	ML110280229	G20110047/LTR-11-0021/EDATS: SECY-2011-0029 - Ltr. to Said Abdel-Khalik re: Report on the Safety Aspects of the Aircraft Impact Assessment for the Westinghouse AP1000 Design Certification Amendment Application.	Borchardt R W	NRC/EDO	EDATS: SECY-2011-0029 G20110047 LTR-11-0021 SECY-2011-0029
02/17/11	ML110490543	Submittal of AP1000 DCD Page Revisions and Associated Confirmatory Item Responses.	Ziesing, R. F.	Westinghouse	DCP_NRC_003132
02/21/11	ML110550390	Westinghouse, Update to Wording in DCD Chapter 15 on Offsite Doses.	Ziesing, R. F.	Westinghouse	DCP_NRC_003137
02/21/11	ML110550391	Westinghouse, Wording Changes on ITAAC Inspectability Concerns and Editorial Corrections.	Ziesing, R. F.	Westinghouse	DCP_NRC_003136

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02/21/11	ML110550455	Westinghouse, Response to Proposed Confirmatory Item in Support of AP1000 Design Certification Amendment Application.	Ziesing, R. F.	Westinghouse	DCP_NRC_003134
02/21/11	ML110550456	Westinghouse, Proprietary Review for Portions of DCD Chapter 3 NRC Comments.	Ziesing, R. F.	Westinghouse	DCP_NRC_003140
02/22/11	ML110550594	Westinghouse, Editorial Correction to Remove Trade Name from DCD Table 6.2.1.1-8 for Rev. 19.	Ziesing, R. F.	Westinghouse	DCP_NRC_003139
02/22/11	ML110550600	Westinghouse, DCD Introduction Wording Update for Registration of AP1000 Trademark.	Ziesing, R. F.	Westinghouse	DCP_NRC_003141
02/23/11	ML110590455	AP1000 Containment Cleanliness - DCD Markup for Rev. 19.	Ziesing, R. F.	Westinghouse	DCP_NRC_003113
02/23/11	ML110590480	Submittal of WCAP-16674, Revision 4, "AP1000 I&C Data Communication and Manual Control of Safety Systems and Components."	Ziesing, R. F.	Westinghouse	DCP_NRC_003115
02/24/11	ML110480631	01/10/11 - 01/14/11 Summary of Meeting with Westinghouse Regarding Tier 2* Items in the AP1000 Design Control Document, held in Cranberry Township, PA.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	
02/24/11	ML110620129	Submittal of APP-GW-GLR-137, Revision 1 "Bases of Digital Overpower and Over-Temperature Delta-T Reactor Trips."	Ziesing, R. F.	Westinghouse	DCP_NRC_003117

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02/25/11	ML110601158	WCAP-16361, Revision 1, Westinghouse Setpoint Methodology for Protection Systems - AP1000.	Ziesing, R. F.	Westinghouse	CI-SRP-7.2-ICE-07 DCP_NRC_003129
02/25/11	ML110610685	Westinghouse, AP1000 DCD Tier 2 Chapter 16 Proposed Editorial Changes. Proposed Editorial/Consistency Correction Responses Are Submitted In Support Of The AP1000 Design Certification Amendment Application (Docket No. 52-006).	Ziesing, R. F.	Westinghouse	DCP_NRC_003131
02/25/11	ML110610782	AP1000 RCP Flywheel 18Cr/18Mn Retainer Ring Material SCC Testing Program.	Ziesing, R. F.	Westinghouse	DCP_NRC_003146
02/25/11	ML110620126	Westinghouse, Regarding Responses To NRC Confirmatory Items From Chapter 16 And Chapter 23 In Support Of The AP1000 Design Certification Amendment Application (Docket No. 52-006).	Ziesing, R. F.	Westinghouse	DCP_NRC_003130
02/25/11	ML110670277	CI-SRP-7.A-SRSB-03, Markup of DCD Revision 18, Chapter 15-Section 15.4.2 in Support of the AP1000 Design Certification Amendment Application.	Ziesing, R. F.	Westinghouse	DCP_NRC_003127
02/28/11	ML110630110	Re-submittal of Appendix 19F Security-Related Information.	Ziesing, R. F.	Westinghouse	DCP_NRC_003126 DCP_NRC_003148
02/28/11	ML110670191	WCAP-16438-NP, Revision 3, "FMEA of AP1000 Protection and Safety Monitoring System."	Domitrovich, B.	Westinghouse	AW-11-3069 DCP_NRC_003116

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02/28/11	ML110691050	APP-GW-S2R-010, Revision 5 (TR03) "Extension of Nuclear Island Seismic Analyses to Soil Sites".		Westinghouse	DCP_NRC_003153
02/28/11	ML110691051	APP-GW-GLR-044, Revision 2, (TR85) "Nuclear Island Basemat and Foundation".		Westinghouse	DCP_NRC_003153
03/01/11	ML110630109	Response to Requests for Additional Information on PCS Operation.	Ziesing, R. F.	Westinghouse	DCP_NRC_003147
03/01/11	ML110630111	Westinghouse Preparation for Engineering Design Verification (EDV) Inspection.	Ziesing, R. F.	Westinghouse	DCP_NRC_003150
03/02/11	ML110630108	OI-SRP7.2-ICE-08-WCAP-16361, Revision 1.	Ziesing, R. F.	Westinghouse	DCP_NRC_003149 WCAP-16361, Rev 1
03/02/11	ML110670188	CI-SRP-7.2-ICE-02, CI-SRP7.3-ICE-2, WCAP-16438, Revision 3, "FMEA of AP1000 Protection and Safety Monitoring System."	Ziesing, R. F.	Westinghouse	AW-11-3069 DCP_NRC_003116 WCAP-16438, Rev 3
03/03/11	ML110670233	CI-SRP16-CTSB-38; Spent Fuel Rack Dimensions; Section 9.1.2.2.1 and Section 16.	Ziesing, R. F.	Westinghouse	AW-11-3095 CI-SRP16-CTSB-38 DCP_NRC_003133
03/03/11	ML110670296	APP-GW-GLR-029NP Rev. 3, "AP1000 Spent Fuel Storage Racks Criticality Analysis," Non-Proprietary, Enclosure 4 to DCP_NRC_003138.	Anton, S.	Holtec International	AW-11-3096 DCP_NRC_003138
03/03/11	ML110670298	Submittal of Spent Fuel Storage Racks Criticality Analysis, APP-GW-GLR-029 (TR65), Rev. 3.	Ziesing, R. F.	Westinghouse	APP-GW-GLR-029, Rev. 3 AW-11-3096 DCP_NRC_003138 TR65

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03/07/11	ML110670311	Advanced submittal of the DCD Revision 19, Redacted Public Version of Appendix 19F.	Ziesing, R. F.	Westinghouse	DCP_NRC_003152
03/08/11	ML110670206	Explanatory Note for Appendix 19F (Aircraft Impact Assessment Information).	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	
03/08/11	ML110691049	Transmittal of Technical Reports 03, 85 and 115.	Ziesing, R. F.	Westinghouse	DCP_NRC_003153
03/10/11	ML110700167	03/10/2011 Functional Arrangement Tier 1 Definition.	Cerne, T. Chapman, T. A. Welch, C. R.	NRC/NRO NRC/RGN-I NRC/RGN-II	
03/10/11	ML110760228	Transmittal of Technical Report APP-GW-GLR-096 Revision 2, (Proprietary) and APP-GW-GLR-097 Revision 2 (Non-Proprietary) "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses".	Ziesing, R. F.	Westinghouse	DCP_NRC_003128 FOIA/PA-2011-0176
03/10/11	ML110760230	APP-GW-GLR-097, Revision 2, "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses," Enclosure 4 to DCP_NRC_003128.		Westinghouse	DCP_NRC_003128
03/15/11	ML110770282	10 CFR 50.46 Annual Report for the AP1000 Standard Plant Design.	Ziesing, R. F.	Westinghouse	DCP_NRC_003144
03/21/11	ML110740411	Audit Report for AP1000 Design Certification Amendment Spent Fuel Rack Dimension Changes - January 31, 2011.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	

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03/23/11	ML110760223	AP1000 Request For Withholding Information From Public Disclosure, Evaluation Of The Effect Of AP1000 Enhanced Shield Building Design On The Containment Response And Safety Analyses (DCP_NRC_003128).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003128 FOIA/PA-2011-0176
03/24/11	ML110840455	2011/03/24 AP-1000 DCD Review - RE: AP1000 Final SE Chapter 23 - Concurrence		NRC/NRO	
03/24/11	ML110890434	HOLD Updates to DCD Introduction to Tier 2 Information.	Ziesing, R. F.	Westinghouse	DCP_NRC_003151
03/28/11	ML110910540	Submittal of APP-GW-GLR-602 (Proprietary) and APP-GW-GLR-603 (Non-Proprietary), "AP1000 Shield Building Design Details for Select Wall and RC/SC Connections".	Ziesing, R. F.	Westinghouse	APP-GW-GLR-602 APP-GW-GLR-603 AW-11-3097 DCP_NRC_003154
03/29/11	ML110900542	Transmittal of Technical Report APP-GW-GLR-044, Revision 3 (TR-85).	Ziesing, R. F.	Westinghouse	DCP_NRC_003159 TR-85
03/30/11	ML110820794	AP1000 Request for Withholding Information From Public Disclosure, CI-SRP-7.2-ICE-07-WCAP-16 361, Revision 1 (DCP_NRC_003129).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003129
03/30/11	ML110900544	Letter Withdrawal of Westinghouse Letter DCP_NRC_003142, Dated 03/24/2011, in Support of the AP1000 Design Certification Amendment Application.	Ziesing, R. F.	Westinghouse	DCP_NRC_003142 DCP_NRC_003160

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03/31/11	ML110910543	Transmittal of APP-GW-GLR-005, Rev. 5 "Containment Vessel Design Adjacent to Large Penetrations," (TR09).	Ziesing, R. F.	Westinghouse	DCP_NRC_003158
04/01/11	ML110250634	06/23/2010 - 09/02/2010 Summary of the AP1000 Design Certification - Regulatory On-Site and Off-Site Reviews of Open Items for the WESTEMS Computer Code (Redacted Version).	Clark, P.	NRC/NRO/DNRL/ NWE2	
04/01/11	ML110910419	2011/04/01 AP-1000 DCD Review - RE: AP1000 Final SE Chapter 9 - Concurrence		NRC/NRO	
04/04/11	ML110871707	AP1000 Request For Withholding Information From Public Disclosure, CI-SRP-7.2-Ice-02, CI-SRP7.3-ICE-02, WCAP-16438, Revision 3, "FMEA Of AP1000 Protection And Safety Monitoring System" (DCP_NRC_003116).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003116 WCAP-16438, Rev 3
04/04/11	ML110960573	Westinghouse, Response to NRC Comments on Proprietary Review of WESTEMS Audit Report.	Ziesing, R. F.	Westinghouse	DCP_NRC_003157
04/06/11	ML110980610	Wording Change for DCD Chapter 2 Editorial Correction.	Ziesing, R. F.	Westinghouse	DCP_NRC_003163

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04/08/11	ML110871927	AP1000 Request For Withholding Information From Public Disclosure, CI-SRP-7.2-ICE-06, WCAP-16674, Revision 4, "AP1000 I&C Data Communication And Manual Control Of Safety Systems And Components" (DCP_NRC_003115)	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003115 WCAP-16674, Rev 4
04/12/11	ML111030427	2011/04/12 AP-1000 DCD Review - AP1000 Final SE Chapter 3 - Concurrence		NRC/NRO	
04/13/11	ML111020005	04/12/2011 Notice of Closed Meeting with Westinghouse to Discuss the AP1000 Shield Building Design Methodology.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	
04/15/11	ML110950197	AP1000 Request For Withholding Information From Public Disclosure, CI-SRP16-CTSB-38; Spent Fuel Rack Dimensions; Section 9.1.2.2.1 and Section 16 (DCP_NRC_003133).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
04/27/11	ML110910315	AP1000 Request for Withholding Information from Public Disclosure, Submittal of Spent Fuel Storage Racks Criticality Analysis, APP-GW-GLR-029 (TR 65) Rev. 3 (DCP_NRC_003138).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	APP-GW-GLR-029, Rev 3
05/03/11	ML110871066	05/03/11 Memorandum Regarding Advisory Committee On Reactor Safeguards Review of AP1000 Design Certification Amendment.	McKenna, E. M.	NRC/NRO/DNRL/ NWE2	

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05/03/11	ML11129A099	Westinghouse Transmittal of Chapter 3 AFSE Proprietary Information Review.	Ziesing, R. F.	Westinghouse	AW-11-3126 DCP_NRC_003167
05/04/11	ML111230096	05/17/11 Notice of Meeting with Westinghouse to Discuss AP1000 Shield Building Design Methodology.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	
05/09/11	ML111250111	AP1000 Request For Withholding Information From Public Disclosure, Transmittal Of Proprietary And Non-Proprietary Reports Related To Turbine Reliability (DCP/NRC1889).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP/NRC1889
05/11/11	ML111230115	AP1000 Request For Withholding Information From Public Disclosure, AP1000 Response To Open Item OI-SRP5.4.1-SRSB-01 Proprietary And Non-Proprietary (DCP_NRC_002838)	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002838
05/16/11	ML11144A068	Presentation Slides "AP1000 Shield Building Design," Meeting with NRC Staff, May 17, 2011 (Proprietary and Non-Proprietary).	Ziesing, R. F.	Westinghouse	AW-11-3159 DCP_NRC_003169
05/16/11	ML11168A096	Presentation Slides "AP1000 Shield Building Design," Meeting with NRC Staff, April 12, 2011.	Ziesing, R. F.	Westinghouse	AW-11-3122 DCP_NRC_003155
05/17/11	ML111190157	04/12-13/2011 Summary of a Category 1 Closed Meeting with Westinghouse Regarding AP1000 Shield Building Design Methodology, in Rockville, Maryland.	Gleaves, W. C.	NRC/NRO/DNRL/ NWE2	

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05/18/11	ML111260315	AP1000 Request For Withholding Information From Public Disclosure, Transmittal Of Proprietary And Non-Proprietary Reports Related To Turbine Reliability (DCP_NRC_002549).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_002549
05/18/11	ML111300435	Response to Requests for Withholding of Proprietary Information in Accordance With 10 CFR Part 2, Section 2.390, "Chapter 3 AFSE Proprietary Information Review"(DCP_NRC_003167).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003167
05/18/11	ML111370357	Follow-Up Inspection of the AP1000 Pressurize Water Reactor Design Aircraft Impact Assessment.	Peralta J D	NRC/NRO/DCIP/CQVB	
05/19/11	ML111370664	06/02/2011 Notice of Meeting with Westinghouse on The AP1000 Containment Vessel Peak Pressure.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
05/19/11	ML11144A188	G20110374/EDATS: OEDO-2011-0366 - Said Abdel-Khalik, Ltr. re: Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document.	Abdel-Khalik S I	NRC/ACRS	EDATS: OEDO-2011-0366 G20110374 OEDO-2011-0366
05/26/11	ML111430582	Summary of A Category 1 Meeting With Westinghouse Regarding AP1000 Shield Building Design Methodology, In Rockville, Maryland On May 17, 2011.	Gleaves W C	NRC/NRO/DNRL/ NWE2	

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06/01/11	ML111470515	Audit Plan for Review of the AP1000 Containment Vessel Pressure Documents - May 23 - June 24, 2011.	Buckberg, P. H.	NRC/NRO/DNRL/ NWE2	
06/06/11	ML111460089	Response to Requests for Withholding of Proprietary Information in Accordance With 10 CFR Part 2, Section 2.390, Presentation Slides "AP1000 Shield Building Design," Meeting With NRC Staff, April 12, 2011 (DCP_NRC_003155).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	
06/06/11	ML111460163	Response To Requests For Withholding Of Proprietary Information In Accordance With 10 CFR Part 2, Section 2.390, Presentation Slides "AP1000 Shield Building Design," Meeting With NRC Staff, May 17, 2011 (DCP_NRC_003169).	Mitra, S. K.	NRC/NRO/DNRL/ NWE2	DCP_NRC_003169
06/06/11	ML11159A035	Westinghouse Re-Submittal of Chapter 7 AFSE Review.	Ziesing, R. F.	Westinghouse	AW-11-3168 DCP_NRC_003172
06/06/11	ML11159A233	Westinghouse, AP1000 Updated Response to RAI OI-SRP5.4.1-CIB1-01 R1.	Ziesing, R. F.	Westinghouse	DCP_NRC_003170 WEM_DCP_000593
06/08/11	ML11164A079	Presentation Slides "AP1000 Design Control Document - Containment Pressure Analysis," Meeting with NRC Staff, June 2, 2011.	Ziesing, R. F.	Westinghouse	DCP_NRC_003174
06/12/11	ML11166A214	Wording Change for DCD Introduction, References and Chapter 7 Editorial Correction	Ziesing, R. F.	Westinghouse	DCP_NRC_003165

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06/12/11	ML11166A215	Transmittal of Revision 19 DCD Markups Resulting from the Revision of APP-GW-GLR-096 Proprietary), "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses" for the Updated Peak.....	Ziesing, R. F.	Westinghouse	DCP_NRC_003171
06/12/11	ML11168A044	DCD Updates to Chapter 3 Information.	Ziesing, R. F.	Westinghouse	DCP_NRC_003161
06/12/11	ML11168A047	Enclosure 6 - Markup of DCD Revision 18, Section 3.7 and Appendix 3H figures that include SUNSI.		Westinghouse	DCP_NRC_003161
06/12/11	ML11180A087	Enclosures 3, 4, 5, 8 and 9 re Markup of DCD, Revision 18.	Ziesing, R. F.	Westinghouse	DCP_NRC_003161
06/13/11	ML11171A301	Westinghouse AP1000 Cover Letter Rev. 19 - Revision 19 Transmittal Letter	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.CVR.P APP-GW-GL-700.CVR.P.19 Submission 11
06/13/11	ML11171A303	Westinghouse AP1000 Design Control Document Rev. 19 - Introduction	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A304	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 - Change Roadmap	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A305	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 - List of Effective Pages	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A306	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A307	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 1 - Introduction - Section 1.0 Introduction	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A308	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.1 Reactor	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A310	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.2 Nuclear Safety Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A311	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.3 Auxiliary Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A312	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.4 Steam and Power Conversion Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A313	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.5 Instrumentation and Control Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A315	Westinghouse Updated Application to Amend AP1000 Nuclear Power Plant Design Certification Rule.	Rupprecht, S. D.	Westinghouse	DCP_NRC_003177

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06/13/11	ML11171A314	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.6 Electrical Power Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A316	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 2 - System Based Design Descriptions and ITAAC - 2.7 HVAC Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A317	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.1 Emergency Response Facilities	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A318	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.2 Human Factors Engineering	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A319	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.3 Buildings	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A320	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.4 Initial Test Program	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A321	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.5 Radiation Monitoring	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A322	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.6 Reactor Coolant Pressure Boundary Leak Detection	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A323	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 3 - Non-System Based Design Descriptions and ITAAC - 3.7 Design Reliability Assurance Program	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A324	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 4 - Interface Requirements - Section 4.0 Interface Requirements	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A325	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 1 Chapter 5 - Site Parameters - 5.0 Site Parameters	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A326	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 - List of Effective Pages	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A327	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 - Change Pages	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A328	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 - Master Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A329	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.1 Introduction	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A330	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.2 General Plant Description	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A331	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.3 Comparisons With Similar Facility Designs	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A332	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.4 Identification of Agents and Contractors	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A333	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.5 Requirements for Further Technical Information	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A334	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.6 Material Referenced	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A335	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.7 Drawings and Other Detailed Information	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A336	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.8 Interfaces for Standard Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A337	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Section 1.9 Compliance with Regulatory Criteria	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A338	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A339	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Appendix 1A Conformance With Regulatory Guides	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A340	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 1 - Introduction and General Description of the Plant - Appendix 1B Severe Accident Mitigation Design Alternatives	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A341	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 10 – Steam and Power Conversion System – Section 10.1 Summary Description	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A342	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 10 – Steam and Power Conversion System – Section 10.2 Turbine-Generator	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A343	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 10 – Steam and Power Conversion System – Section 10.3 Main Steam Supply System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A344	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 10 – Steam and Power Conversion System – Section 10.4 Other Features of Steam and Power Conversion System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A345	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 10 – Steam and Power Conversion System – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A346	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 11 – Radioactive Waste Management – Section 11.1 Source Terms	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A347	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 11 – Radioactive Waste Management – Section 11.2 Liquid Waste Management Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A348	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 11 – Radioactive Waste Management – Section 11.3 Gaseous Waste Management System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A349	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 11 – Radioactive Waste Management – Section 11.4 Solid Waste Management	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A350	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 11 – Radioactive Waste Management – Section 11.5 Radiation Monitoring	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A351	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 11 – Radioactive Waste Management – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A352	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 12 – Radiation Protection – Section 12.1 Assuring that Occupational Radiation Exposures Are As-Low-As-Reasonably Achievable (ALARA)	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A353	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 12 – Radiation Protection – Section 12.2 Radiation Sources	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A354	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 12 – Radiation Protection – Section 12.3 Radiation Protection Design Feature	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A355	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 12 – Radiation Protection – Section 12.4 Dose Assessment	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A356	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 12 – Radiation Protection – Section 12.5 Health Physics Facilities Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A357	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 12 – Radiation Protection – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A358	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 13 - Conduct of Operations - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A359	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 13 - Conduct of Operations	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A360	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 14 - Initial Test Program - Section 14.1 Specific Information to be Included in Preliminary Final Safety Analysis Reports	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A362	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 14 - Initial Test Program - Section 14.2 Specific Information to be Included in Standard Safety Analysis Reports	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A363	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 14 - Initial Test Program - Section 14.3 Certified Design Material	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A364	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 14 - Initial Test Program - Section 14.4 Combined License Applicant Responsibilities	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A365	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 14 - Initial Test Program - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A366	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 14 - Appendix 14A DAC ITAAC Closure Process	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A367	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.0 Accident Analyses	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A368	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.1 Increase in Heat Removal From the Primary System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A369	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.2 Decrease in Heat Removal by the Secondary System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A370	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.3 Decrease in Reactor Coolant System Flow Rate	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A371	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.4 Reactivity and Power Distribution Anomalies	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A372	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.5 Increase in Reactor Coolant Inventory	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A373	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.6 Decrease in Reactor Coolant Inventory	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A374	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.7 Radioactive Release from a Subsystem or Component	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A375	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Section 15.8 Anticipated Transients Without Scram	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A377	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A378	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Appendix 15A Evaluation Models And Parameters for Analysis of Radiological Consequences of Accidents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A379	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 15 - Accident Analyses - Appendix 15B Removal of Airborne Activity from the Containment Atmosphere Following a LOCA	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A380	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 16 - Technical Specifications - 16.1 Bases Part 1	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A381	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 16 - Technical Specifications - 16.1 Bases Part 2	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A382	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 16 - Technical Specifications - 16.1 Technical Specifications	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A383	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 16 - Technical Specifications - 16.2 Design Reliability Assurance Program	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A384	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 16 - Technical Specifications - 16.3 Investment Protection	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A385	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 16 - Technical Specifications - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A386	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 17 - Quality Assurance - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A387	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 17 - Quality Assurance	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A388	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.10 Training Program Development	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A389	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.11 Human Factors Engineering Verification and Validation	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A390	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.12 Inventory	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A391	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.13 Design Implementation	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A392	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.14 Human Performance Monitoring	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A393	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.1 Overview	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A394	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.2 Human Factors Engineering Program Management	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A395	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.3 Operating Experience Review	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A396	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.4 Functional Requirements Analysis and Allocation	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A397	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.5 AP1000 Task Analysis Implementation Plan	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A398	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.6 Staffing	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A399	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.7 Integration of Human Reliability Analysis with Human Factors Engineering	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A400	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.8 Human System Interface Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A401	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Section 18.9 Procedure Development	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A402	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 18 - Human Factors Engineering - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A403	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19-1 to 19.14	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A404	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.15 to 19.33	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A405	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.34 to 19.35	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A406	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.36 to 19.38	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A408	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.39 to 19.40	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A409	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.41 to 19.54	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A410	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.55 to 19.58	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A411	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Sections 19.59 PRA Results and Insights	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A412	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A413	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Appendix 19A Thermal Hydraulic Analysis To Support Success Criteria	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A414	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Appendix 19B Ex-Vessel Severe Accident Phenomena	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A415	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Appendix 19C Additional Assessment of AP1000 Design Features	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A416	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Appendix 19D Equipment Survivability Assessment	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A417	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Appendix 19E Shutdown Evaluation	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A418	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 19 - Probabilistic Risk Assessment - Appendix 19F Malevolent Aircraft Impact	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 SUBMISSION 11
06/13/11	ML11171A419	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 2 - Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A420	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 2 - Site Characteristics	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 WESTINGHOUSE SUBMISSION 11

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06/13/11	ML11171A422	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A423	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.11 Environmental Qualification of Mechanical and Electrical Equipment	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A424	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.1 Conformance with Nuclear Regulatory Commission General Design Criteria	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A425	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.2 Classification of Structures, Components, and Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A426	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.3 Wind and Tornado Loadings	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A427	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.4 Water Level (Flood) Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A428	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.5 Missile Protection	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A429	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.6 Protection Against the Dynamic Effects Associated with the Postulated	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A430	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.7 Seismic Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A431	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.8 Design of Category I Structures	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A432	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Section 3.9 Mechanical Systems and Components	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A433	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A434	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3A HVAC Ducts And Duct Supports	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A435	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3B Leak-Before-Break Evaluation Of The AP1000 Piping	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A436	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3C Reactor Coolant Loop Analysis Methods	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A437	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3D Methodology For Qualifying AP1000 Safety-Related Electrical And M	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A438	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3E High-Energy Piping in The Nuclear Island	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A439	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3F Cable Trays and Cable Tray Supports	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A440	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3G Nuclear Island Seismic Analyses	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A441	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3H Auxiliary and Shield Building Critical Sections	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A442	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 3 - Design of Structures, Components, Equip. and Systems - Appendix 3I Evaluation for High Frequency Seismic Input	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A443	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Section 4.1 Summary Description	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A444	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Section 4.2 Fuel System Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A445	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Section 4.3 Nuclear Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A446	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Section 4.4 Thermal and Hydraulic Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A447	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Section 4.5 Reactor Materials	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A448	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Section 4.6 Functional Design of Reactivity Control Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A449	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 4 – Reactor – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A450	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 5 – Reactor Coolant System and Connected Systems – Section 5.1 Summary Description	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A451	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 5 – Reactor Coolant System and Connected Systems – Section 5.2 Integrity of Reactor Coolant Pressure Boundary	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A453	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 5 – Reactor Coolant System and Connected Systems – Section 5.3 Reactor Vessel	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A454	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 5 – Reactor Coolant System and Connected Systems – Section 5.4 Component and Subsystem Design	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A455	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 5 – Reactor Coolant System and Connected Systems – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A456	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 - Engineered Safety Features - Section 6.0 Engineered Safety Features	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A457	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Section 6.1 Engineered Safety Features Materials	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A458	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Section 6.2 Containment Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A459	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Section 6.3 Passive Core Cooling System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A460	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Section 6.4 Habitability Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A461	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Section 6.5 Fission Product Removal and Control Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A462	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Section 6.6 Inservice Inspection of Class 2 and 3 Components	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A463	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 – Engineered Safety Features – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A464	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 6 - Engineered Safety Features - Appendix 6A Fission Product Distribution in the Post-DBA Containment Atmosphere	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A465	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.1 Introduction	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A466	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.2 Reactor Trip	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A467	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.3 Engineered Safety Features	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A468	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.4 Systems Required for Safe Shutdown	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A469	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.5 Safety-Related Display Information	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A470	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.6 Interlock Systems Important to Safety	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A471	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls - Section 7.7 Control and Instrumentation Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A472	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 7 - Instrumentation and Controls – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A474	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 8 – Electric Power – Section 8.1 Introduction	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A478	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 8 – Electric Power – Section 8.2 Offsite Power System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A483	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 8 – Electric Power – Section 8.3 Onsite Power Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A487	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 8 – Electric Power – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A491	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Section 9.1 Fuel Storage and Handling	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A493	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Section 9.2 Water Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A494	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Section 9.3 Process Auxiliaries	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A495	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Section 9.4 Air-Conditioning, Heating, Cooling, and Ventilation System	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11

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06/13/11	ML11171A497	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Section 9.5 Other Auxiliary Systems	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A498	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Table of Contents	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11171A499	Westinghouse AP1000 Design Control Document Rev. 19 - Tier 2 Chapter 9 – Auxiliary Systems – Appendix 9A Fire Protection Analysis	Rupprecht, S. D.	Westinghouse	APP-GW-GL-700.DCD.P APP-GW-GL-700.DCD.P.19 Submission 11
06/13/11	ML11173A143	Westinghouse, Submittal of "AP1000 NRC Instructor Training Course."	Gresham J A	Westinghouse	LTR-NRC-11-28
06/14/11	ML111640175	06/30/2011-Notice of Public Meeting With Westinghouse On The AP1000 Design Certification - Shield Building Roof Passive Containment Cooling Water Storage Tank Analysis.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	
06/14/11	ML111640220	06/30/2011-Notice of Public Meeting With Westinghouse on The AP1000 Design Certification - Presentation of Revision 19.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	
06/14/11	ML11168A009	Submittal of APP-GW-GLR-602 Revision 1 (Proprietary) and APP-GW-GLR-603 Revision 1 (Non-Proprietary) "AP1000 Shield Building Design Details for Select Wall and RC/SC Connections."	Ziesing, R.	Westinghouse	APP-GW-GLR-603, Rev 1 DCP_NRC_003162

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06/14/11	ML11168A040	Transmittal of Technical Report APP-GW-GLR-096, Revision 3 (Proprietary) and APP-GW-GLR-097, Revision 3 (Non-Proprietary) "Evaluation of the Effect of the AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analyses".	Ziesing, R. F.	Westinghouse	APP-GW-GLR-096, Rev 3 APP-GW-GLR-097, Rev 3 DCP_NRC_0003178
06/16/11	ML111662183	06/30/2011-Notice of Public Meeting With Westinghouse On The AP1000 Design Certification - Shield Building Roof Passive Containment Cooling Water Storage Tank Analysis.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	
06/16/11	ML111662209	06/30/2011-Notice of Public Meeting With Westinghouse On The AP1000 Design Certification - Presentation Of Revision 19.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	
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06/21/11	ML111721096	06/30/11 Revised Notice of Meeting with Westinghouse on The AP1000 Design Certification - Presentation of Revision 19.	Gleaves, B.	NRC/NRO/DNRL/ NWE2	

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C. ABBREVIATIONS

χ/Q	atmospheric dispersion
µm	micrometer
µm/m	micrometers per meter
µSv/h	microSieverts per hour
2D	two dimensional
2oo2	two-out-of-two
2oo3	two-out-of-three
2oo4	two-out-of-four
3D	three dimensional
ac	alternating current
ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AEA	Atomic Energy Act
AF	Advant Fieldbus
AFCAP	Advanced First Core Analysis Program
AFSE	Advanced Final Safety Evaluation
Ag-In-Cd	silver-indium-cadmium
AIA	aircraft impact assessment
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as is reasonably achievable
ALI	annual limit on intake
AlOOH	aluminum oxyhydroxide
ALWR	advanced light-water reactor
AMS	Asset Management Solutions
AMS	Aerospace Material Specification
AMSAC	anticipated transient without scram (ATWS) mitigation systems actuation circuitry
ANS	American National Standard
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOA	axial offset anomaly
AOI	Advant-Ovation Interface
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
AOV	air operated valve
API	American Petroleum Institute
ARP	Alarm Response Procedures
ART	adjusted reference temperature
ASB	auxiliary and shield building
ASCE	American Society of Civil Engineers
ASD	allowable stress design
ASHRAE	American Society of Heating, Refrigerating and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers

ASTM	American Society for Testing and Materials
ASTRUM	Automated Statistical Treatment of Uncertainty Method
ATWS	anticipated transients without scram
AV	allowable values
AWS	American Welding Society
B&PV	Boiler & Pressure Vessel (ASME B&PV Code)
B&W	Babcock & Wilcox
BAC	bounding analysis curve
BDBA	beyond design basis accident
BELOCA	best estimate large-break loss-of-coolant accident
BPL	bistable processor logic
BTP	Branch Technical Position
Btu	British thermal units
BTU/ft-hr-°F	British thermal unit per foot hour degree Fahrenheit
BWR	boiling-water reactor
C	Celsius
$Ca(PO_4)_2$	calcium phosphate
CAFTA	Computer-Aided Fault-Tree Analysis System
Cal-Sil	calcium silicate
CAS	central alarm station
CASS	cast austenitic stainless steel
CBP	computer-based procedure
CCF	common-cause failure
CCS	component cooling water system
CCW	component cooling water
CCWS	component cooling water system
CDF	core damage frequency
CDF	cumulative distribution function
CDFM	conservative deterministic failure margin
CDI	conceptual design information
CE	Combustion Engineering
CEI/IEC	Commission Electrotechnique Internationale/International Electrotechnical Commission
CET	core exit thermocouple
CEUS	Central and Eastern United States
CFD	computational fluid dynamics
cfm	cubic feet per minute
CFR	*Code of Federal Regulations*
CFS	condensate and feedwater system
CHF	critical heat flux
CIM	component interface module
CIPS	crud-induced power shift
CIS	containment internal structure
CIS	containment isolation system
CL	cold leg
CLP	cask loading pit
cm	centimeter
cm^2	square centimeter
CMT	core makeup tank

COF	coefficient of friction
COL	combined license
COLA	combined license application
COLR	core operating limits report
CP	cathodic protection
CP	construction permit
CPS	Computerized Procedure System
CPU	central processing unit
CQD	Code Qualification Document
CR	control room
CRC	cyclic redundancy check
CRD	control rod drive
CRDM	control rod drive mechanism
CRDS	control rod drive system
CRE	control room envelope
Cr-Mo	chromium-molybdenum
CSA	control support area
CSDRS	certified seismic design response spectra
CTS	custom technical specification
Cu	copper
CV	containment vessel
CV	control valve
CVS	chemical and volume control system
CWO	core-wide oxidation
CWS	circulating water system
DAC	derived air concentration
DAC	design acceptance criteria
DAS	diverse actuation system
DBA	design-basis accident
DBE	design-basis event
DBT	design-basis threat
DC	design certification
dc	direct current
DCD	design control document
DCP	design change proposal
DCP	design change package
DCR	design certification rule
DDS	data display and processing system
DECL	double-ended cold-leg
DECLG	double-ended cold-leg guillotine
DEDVI	double-ended direct vessel injection
D-EHC	digital electrohydraulic control
DEHL	double-ended hot-leg
DFC	diagnostic flow chart
DG	diesel generator
DG	draft guide
DI&C	digital instrumentation and controls
DID	defense-in-depth
DMIMS	digital metal impact monitoring system
DNB	departure from nucleate boiling

DNBR	departure from nuclear boiling ratio
DOF	degree-of-freedom
dP	differential pressure
dpa	displacements per atom
D-RAP	Design Reliability Assurance Program
DVI	direct vessel injection
DWTP	double wide transition panel
EAB	exclusion area boundary
EALF	energy of the average lethargy causing fission
EC	composite cored/stranded electrodes
ECCS	emergency core cooling system
EDF	Electricite de France
EFDS	equipment and floor drainage system
EFPY	effective full-power year
EHX	external heat exchanger
El.	Elevation
EMC	electromagnetic compatibility
ENDF	Evaluated Nuclear Data File
EO	equipment operator
EOF	Emergency Operating Facility
EOL	end-of-life
EOP	emergency operating procedure
EP	expert panel
EPRI	Electric Power Research Institute
EQ	strip electrodes
EQ	environmental qualification
ERO	emergency response organization
ERS	envelope response spectra
ERVC	external reactor vessel cooling
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESP	early site permit
F	Fahrenheit
FAC	flow-accelerated corrosion
FCEP	fuel criterion evaluation process
FEA	finite element analysis
FEM	finite element method
FEMA	Federal Emergency Management Agency
FFT	Fast-Fourier Transforms
FHA	fuel handling area
FHA	fuel-handling accident
FHM	fuel handling machine
FIRS	foundation input response spectra
FIV	flow-induced vibration
FME	foreign material exclusion
FMEA	failure modes and effects analysis
FN	ferrite number
FPDS	flat panel display system
FPGA	field programmable gate array

Fps	feet per second
FPS	fire protection system
FR	firm rock
FR	*Federal Register*
FRN	*Federal Register* Notice
FRS	floor response spectra
FSAR	final safety analysis report
FSER	final safety evaluation report
ft	feet
ft/s	feet per second
ft^2	square feet
ft^3	cubic feet
ft^3/min	cubic feet per minute
g/cm^3	grams per cubic centimeters
GDC	General Design Criteria
GL	Generic Letter
GMAW	gas metal arc welding
GMRS	ground motion response spectra
GOI	generic open item
GOP	general operating procedure
gpm	gallons per minute
gpm/ft^2	gallons per minute per square foot
GRCA	gray rod cluster assembly
GSI	generic safety issue
GTS	generic technical specification
H_2	hydrogen
HA	human action
HAZ	heat affected zone
HCLPF	high confidence in low probability of failure
HDCI	high-duty core index
HDPE	high-density polyethylene
HED	human engineering discrepancy
HEPA	high-efficiency particulate air
HFE	human factors engineering
HFT	hot functional test
HL	hot leg
HMR	hydro-meteorological report
HP	health physics
HR	hard-rock
HRA	human reliability analysis
HRHF	hard rock high frequency
HSI	human-system interface
HSL	high speed link
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
Hz	Hertz
I&C	instrumentation and control
I/O	input/output

IASCC	irradiation-assisted stress-corrosion cracking
ICET	integrated chemical effect test
ICP	integrated communications processor
ICTN	integral clamp top nozzle
ID	inner diameter
IDI	isolated development infrastructure
IEEE	Institute of Electrical and Electronic Engineers
IFM	intermediate flow mixer
IGSCC	intergranular stress-corrosion cracking
IHP	integrated head package
IIS	in-core instrumentation system
IISS	in-core instrumentation-support structure
IITA	in-core instrument thimble assembly
ILP	integrated logic processor
in	inch
IN	Information Notice
in/h	inch per hour
in/ft	inch per foot
in/in	inch per inch
in^2	square inch(es)
IOZ	inorganic zinc
IPEEE	individual plant examination of external events
IPSAC	Investment Protection Short-Term Availability Control
IRP	important by the regulatory treatment of nonsafety systems (RTNSS) process
IRWST	in-containment refueling water storage tank
ISA	International Society of Automation
ISG	Interim Staff Guidance
ISI	inservice inspection
ISLOCA	intersystem loss-of-coolant accident
ISRS	in-structure response spectra
IST	inservice testing
ISV	integrated system validation
ITAAC	inspection, tests, analyses, and acceptance criteria
ITP	interface and test processor
IV&V	independent verification and validation
JOG	Joint Owners Group
kA	kilo amp
kcf	kips per cubic feet
kg	kilograms
kg/h	kilograms per hour
kg/m^2	kilograms per square meter
kg/m^3	kilograms per cubic meter
kg/s	kilograms per second
kip	kilopounds (1000 pounds)
kJ/h	kilojoules per hour
km	kilometers
km/h	kilometers per hour
kPa	kilopascal
ksf	kilopounds per square foot

ksi	kilopounds per square inch
kV	kilovolt
kVA	kilovolt amps
kW	kilowatts
kW/ft	kilowatts per foot
L	liters
L/D	length to diameter ratio
lb	pound
lb/ft^2	pounds per square foot
lb/ft^3	pounds per cubic foot
lb/h	pounds per hour
lb/s	pounds per second
LBB	leak-before-break
lbf	pounds-force
lbm	pounds-mass
lbm/ft^3	pounds-mass per cubic foot
lbm/s	pounds-mass per second
lbs	pounds
LCL	local coincidence logic
LCO	limiting condition for operation
LCS	local control station
LCSP	lower core support plate
LLHS	light load handling system
LOAC	loss of alternating current power
LOCA	loss-of-coolant accident
LOCADM	Loss-of-Coolant Accident Deposition Model
LOFT	loss-of-fluid test
LOSP	loss of offsite power
Lpm	liters per minute
Lpm/m^3	liters per minute per square meter
LPZ	low population zone
LRF	large release frequency
LRFD	load and resistance factor design
LSB	last stage blade
LSSS	limiting safety system setting
LTOP	low-temperature overpressure protection
LWR	light-water reactor
m	meters
m/s	meters per second
m^2	square meter
m^3	cubic meters
m^3/h	cubic meters per hour
Mbit	megabit
MBtu	million British thermal units
MBtu/hr	million British thermal units/hour
MCC	motor control center
MCR	main control room
MCRE	main control room envelope
MEB	moisture extraction blade

MFIV	main feedwater isolation valve
MG	motor-generator
MHS	mechanical handling system
MI	mineral insulated
MLO	maximum local oxidation
mm	millimeters
mm/m	millimeters per meter
MOV	motor-operated valve
MOX	Mixed Oxide
MPa	megapascal
MPa/m^3	megapascal per cubic meter
mph	miles per hour
mrem/h	millirem per hour
MRI	metal reflective insulation
MSE	mechanically stabilized earth
MSIV	main steam isolation valve
MSL	main steam line
MSLB	main steam line break
MSSS	main steam supply system
MSSV	main steam safety valve
MSV	main steam stop valve
MTBF	mean time between failure
MTC	maintenance and test cabinet
MTIS	maintenance, inspection, test, and surveillance
MTP	maintenance and test panel
MUX	multiplexer
MVA	megavolt amp
MVG	mixing vane grid
MW	megawatt
MWDF/MTU	megawatt-days per metric ton of uranium
MWe	megawatts electric
MWh	megawatt-hour
MWt	megawatts thermal
$NaAlSi_3O_8$	sodium aluminum silicate
NACE	National Association of Corrosion Engineers
NDE	nondestructive examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NF	new fuel
NFPA	National Fire Protection Association
NI	nuclear island
NIS	nuclear instrumentation system
NOP	Normal Operating Procedures
NPP	nuclear power plant
NPS	nominal pipe size
NPSH	net positive suction head
NPSHa	net positive suction head available
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission (U.S.)
NRCA	nonradiologically controlled area

NS	non-seismic
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
OCS	operations and control centers system
OCS	operational conditions sampling
OECD	Organization for Economic Cooperation and Development
OER	operating experience review
OHLHS	overhead heavy load handling system
OL	operating license
OM	Operation and Maintenance (ASME OM Code)
OPDMS	Online Power Distribution Monitoring System
OPRAA	operational phase reliability assurance activity
OPΔT	overpower delta-T
O-RAP	operational reliability assurance program
OSA	operational sequence analysis
OTΔT	overtemperature delta-T
P&ID	piping and instrumentation diagram
P&ID	piping and instrumentation drawings
P/F	pass/fail
P/T	pressure-temperature
PA	protected area
Pa	pascal
PABX	private automatic branch exchange
PAMS	post-accident monitoring system
PCCAWST	passive containment cooling ancillary water storage tank
PCCS	passive core cooling system
PCCWST	passive containment cooling water storage tank
PCS	passive containment cooling system
PCT	peak cladding temperature
PDSP	primary dedicated safety panel
PFM	probabilistic fracture mechanics
PGA	peak ground acceleration
PI	plus integral
PLS	plant control system
PLS	plant lighting system
P_m	primary membrane
$P_m + P_b$	primary membrane plus bending
PMF	probable maximum flood
PMP	probable maximum precipitation
PMS	protection and safety monitoring system
PORV	power-operated relief valve
POV	power-operated valve
ppm	parts per million
PRA	probabilistic risk assessment
PRE	pitting resistance equivalent
PRHR	passive residual heat removal
PSAI	plant-specific action item
PSD	power spectral density
psf	pounds per square foot

psi	pounds per square inch
PSI	preservice inspection
psia	pounds per square inch absolute
psid	pounds per square inch differential
psig	pounds per square inch gauge
PTLR	pressure-temperature limits report
PTS	plant-specific technical specification
PVC	plasticized polyvinyl chloride
PWR	pressurized-water reactor
PWS	potable water system
PXS	passive core cooling system
QA	quality assurance
QDPS	qualified data processing system
QG	quality group
QMS	quality management system
RAI	request for additional information
RAP	reliability assurance program
RAT	reserve auxiliary transformer
RAW	risk achievement worth
RC	reinforced concrete
RCCA	rod cluster control assembly
RCDT	reactor coolant drain tank
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDP	refueling disconnect panel
rem	roentgen equivalent man
REMP	Radiological Environmental Monitoring Program
RESAR	Reference Safety Analysis Report
RETS	radiological effluent technical specifications
RG	regulatory guide
RHR	residual heat removal
RIM	required input motion
RIS	Regulatory Issue Summary
RLE	review-level earthquake
RM	refueling machine
RMI	reflective metallic insulation
RMS	radiation monitoring system
RNC	remote node controller
RNS	residual heat removal system
RO	reactor operator
RPV	reactor pressure vessel
RRAS	repair, replacement, and automation services
RRS	required response spectra
RRW	risk reduction worth
RSA	response spectrum analysis
RSC	remote shutdown console
RSR	remote shutdown room

RSS	remote shutdown station
RSW	remote shutdown workstation
RTCB	reactor trip circuit breaker
RTD	resistance temperature detector
RTD	resistance thermowell detector
RTDP	revised thermal design procedure
RTNSS	regulatory treatment of nonsafety systems
RTP	rated thermal power
RT_{PTS}	reference temperature-pressurized thermal shock
RTS	reactor trip system
RV	reactor vessel
RVH	reactor vessel head
RVIS	reactor vessel insulation system
RWS	raw water system
SAFDL	specified acceptable fuel design limit
SAM	startup administrative manual
SAMDA	severe accident mitigation design alternative
SAMG	severe accident management guidance
SAR	safety analysis report
SAS	secondary alarm station
SAT	systems approach to training
SC	steel and concrete composite
SCC	stress-corrosion cracking
scf	standard cubic foot
scfm	standard cubic feet per minute
scmh	standard cubic meters per hour
SCP	setpoint control program
SCV	steel containment vessel
SDOE	secure development and operational environment
SDS	sanitary drainage system
SE	safety evaluation
SEP	Systematic Evaluation Program
SER	safety evaluation report
SF	spent fuel
SFHM	spent fuel pool handling machine
SFHT	spent fuel handling tool
SFP	spent fuel pool
SFS	spent fuel pool cooling system
SFW	startup feedwater system
SG	steam generator
SGI	safeguards information
SGS	steam generator system
SGTR	steam generator tube rupture
SHA	software hazards analysis
SL	safety limit
SLC	software lifecycle
SM	soft-to-medium (soil)
SMA	seismic margin analysis
SMS	special monitoring system
SOE	sequence of events

SPC	steam and power conversion
SPD	self-powered detector
SPHSE	self-priming high-solids epoxy
SPM	Software Program Manual
SPS	signal processing system
SR	soft rock
SR	surveillance requirement
SRM	Staff Requirements Memorandum
SRNC	safety remote node controller
SRP	standard review plan
SRSS	square root of the sum of the squares
SS	soft soil
SS	stainless steel
SSA	signal selector algorithm
SSCs	structures, systems and components
SSE	safe shutdown earthquake
SSI	soil-structure interaction
SSSI	structure-soil-structure interaction
STA	shift technical advisor
STD	Standard
Std.	Standard (IEEE Standard)
STS	standard technical specifications
Sv	sievert
SWCCF	software common-cause failure
SWS	service water system
SWTP	single wide transition panel
T	thickness
TASCS	thermal stratification, cycling or striping
T_{AVG}	average coolant temperature
TC	core inlet temperature signal
TCS	turbine building closed cooling water system
TEDE	total effective dose equivalent
TEM	transmission electron microscopy
TF	transfer functions
TMI	Three Mile Island
TR	technical report
TRS	test response spectra
TS	technical specification
TSC	technical support center
TSP	trisodium phosphate
TSTF	Technical Specification Task Force
UA	heat transfer capacity consisting of the coefficient of heat transfer (U) and required heat transfer area (A)
UAT	unit auxiliary transformer
UBC	Uniform Building Code
UBSM	upper bound soft-to-medium (soil)
UL	Underwriters Laboratories
UMI	upper mounted instrumentations
UNS	unified numbering system

UPS	uninterruptible power supply
URD	Utility Requirements Document
URS	ultimate rupture strength
V	volts
V&V	verification and validation
Vac	volts alternating current
VAS	radiological controlled area ventilation system
VBS	nuclear island nonradioactive ventilation system
VCD	vacuum carbon-deoxidized
VCSNS	Virgil C. Summer Nuclear Station
Vdc	volts direct current
VDU	visual display unit
VES	emergency habitability system
VES	main control room habitability system
VFD	variable frequency drive
VFS	containment air filtration system
VFTP	Ventilation Filter Testing Program
VHS	hot machine shop heating, ventilation, and air conditioning system
VPI	valve position indicator
VRS	radwaste building heating, ventilation, and air conditioning system
Vs	shear wave velocity
VS	void swelling
VTS	turbine building ventilation system
VWS	central chilled water system
VXS	annex/auxiliary buildings nonradioactive heating, ventilation, and air conditioning system
VYS	hot water heating system
VZS	diesel generator building heating and ventilation system
W	watt
WABA	wet annular burnable absorber
WCAP	Westinghouse Commercial Atomic Power
WGS	gaseous radwaste system
WIN	Westinghouse integral nozzle
WLS	liquid radwaste system
WOG	Westinghouse Owners Group
WRS	waste drain system
WSS	solid radwaste system
wt%	weight-percent
WWS	waste water system
ZOI	zone of influence
ZPA	zero period acceleration

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E. WESTINGHOUSE RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Document Date	Accession Number	Title	Case/Reference Number
08/24/06	ML062410492	AP1000 COL Response to Requests for Additional Information (TR #36).	AP1000 DCP/NRC1769 TAC MD2109
09/08/06	ML062560035	AP1000 COL Response to Request for Additional Information (TR #32).	DCP/NRC1772 TAC MD1432
09/15/06	ML062620277	AP1000 COL Response to Requests for Additional Information (TR #37).	AP1000 DCP/NRC1777 TAC MD1433
09/15/06	ML062620350	AP1000 COL Response to Request for Additional Information (TR 6).	DCP/NRC1776
09/22/06	ML062680029	AP1000 COL Response to Requests for Additional Information (TR #36).	DCP/NRC1779
09/27/06	ML062720129	AP1000 COL Response to Requests for Additional Information (TR #6).	
09/29/06	ML062760231	AP1000 COL Response to Requests for Additional Information (TR # 8).	AP1000 APP-GW-GLR-022, Rev. 0 DCP/NRC1786 TAC MD2175
12/12/06	ML063480072	AP1000 COL Response to Request for Additional Information (TR #32).	APP-GW-GLN-002, Rev 0 DCP/NRC1809 RAI-TR32-007 TAC MD1432
12/18/06	ML063540065	AP1000 COL Response to Request for Additional Information (TR #36).	AP1000 DCP/NRC1811
01/29/07	ML070330131	AP1000 COL Response to Request for Additional Information (TR #59).	AP1000 DCP/NRC1819 TAC MD1435
01/29/07	ML070330590	AP1000 COL Response to Request for Additional Information (TR #3).	AP1000 APP-GW-S2R-010, Rev 0 DCP/NRC1822
02/01/07	ML070330598	Westinghouse Electric Co, AP1000 COL Response to Request for Additional Information (TR #6).	AP1000DCP/NRC1823TAC MD2174
02/08/07	ML070430279	AP1000 COL Response to NRC Request for Additional Information (TR#6).	AP1000 APP-GW-GLR-021, Rev 0 DCP/NRC1827
02/08/07	ML070430285	AP1000 COL Response to Request for Additional Information (TR #39).	DCP/NRC1828 TAC MD1849

Document Date	Accession Number	Title	Case/Reference Number
02/16/07	ML070520284	AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-003, Revision 1.	DCP/NRC1818
02/21/07	ML070570085	AP1000 COL Response to Request for Additional Information (TR #36).	DCP/NRC1837
02/21/07	ML070570098	AP1000 COL Response to Request for Additional Information (TR #6).	DCP/NRC1836
02/26/07	ML070610091	AP1000 COL Response to Request for Additional Information (TR #3).	APP-GW-S2R-010, Rev 0 DCP/NRC1840 TAC MD2358
02/27/07	ML070610090	Submittal of AP1000 COL Response to Request for Additional Information (TR #59).	AP1000 DCP/NRC1813
3/9/2007	ML070720534	AP1000 COL Response to Request for Additional Information (TR #3).	DCP/NRC1843 TAC MD2358
03/16/07	ML070810211	AP1000 COL Response to Request for Additional Information (TR #3).	AP1000 DCP/NRC1845 TAC MD2358
03/29/07	ML070930690	Submittal of AP1000 COL Response to Request for Additional Information (TR #3).	DCP/NRC1857 TAC MD2358
04/05/07	ML071010096	AP1000 COL Response to Request for Additional Information (TR #3).	DCP/NRC1858 RAI-TR03-021 RAI-TR03-023
04/09/07	ML071010114	AP1000 COL Response to Request for Additional Information (TR #54).	DCP/NRC1860
04/10/07	ML071010532	AP1000 COL Response to Request for Additional Information (TR #54).	APP-GW-GLR-033, Rev 0 DCP/NRC1861 TAC MD2551
04/13/07	ML071060314	AP1000 COL Response to Request for Additional Information (TR #44).	APP-GW-GLR-026, Rev 0DCP/NRC1866RAI-TR44-01TAC MD2104
04/13/07	ML071060316	AP1000 COL Response to Requests for Additional Information (TR 6).	DCP/NRC1867
04/13/07	ML071070483	AP1000 COL Response to Request for Additional Information (TR #61).	DCP/NRC1868
04/30/07	ML071350571	WCAP-16767-P, Rev. 0, "Response to NRC Request for Additional Information on Westinghouse AP1000 Combined License (COL) Pre-Application Technical Reports Number 42 and Number 88."	APP-PMS-GL-042 DCP/NRC1881

Document Date	Accession Number	Title	Case/Reference Number
05/02/07	ML071270244	Westinghouse Electric Co., AP1000 COL Response to Request for Additional Information (TR #12).	DCP/NRC1874 TAC MD2694
05/03/07	ML071280369	Westinghouse - AP1000 COL Response to Request for Additional Information (TR #44).	DCP/NRC1875
05/04/07	ML071280363	Westinghouse - AP1000 COL Response to Request for Additional Information (TR #71A and 71B).	DCP/NRC1859
05/08/07	ML071290575	AP1000 COL Response to Request for Additional Information (TR #6).	DCP/NRC1880
05/11/07	ML071350099	Submittal of AP1000 COL Response to Requests for Additional Information (TR #43).	APP-GW-GLR-018, Rev 0 AW-07-2279 DCP/NRC1884
05/11/07	ML071350103	Enclosure 3, Response to Requests for Additional Information on Technical Report No. 43, RAI-TR43-001 through RAI-TR43-017.	AP1000 APP-GW-JJ-002 DCP/NRC1884 WCAP-16438
05/11/07	ML071350568	AP1000 COL Response to Request for Additional Information (TR #42 and 88).	DCP/NRC1881 TAC MD1850 TAC MD3831 WCAP-16767-NP, Rev 0 WCAP-16767-P, Rev 0
05/11/07	ML071350570	WCAP-16767-NP, Rev. 0, "Response to Request for Additional Information on Westinghouse AP1000 Combined License (COL) Pre-Application Technical Reports Number 42 and Number 88."	APP-PMS-GL-042DCP/ NRC1881
05/17/07	ML071410075	AP1000 COL Response to Request for Additional Information (TR #54).	APP-GW-GLR-033, Rev 0 DCP/NRC1890
05/17/07	ML071410145	AP1000 COL Response to Request for Additional Information (TR #54).	DCP/NRC1891 TAC MD2551
05/17/07	ML071410146	Enclosure 2, "Response to Request for Additional Information on Technical Report No. 54."	DCP/NRC1891 RAI-TR54-021
05/30/07	ML071520070	AP1000 COL Response to Requests for Additional Information (TR #43) Non-Proprietary Responses.	AP1000 APP-GW-GLR-018, Rev 0 DCP/NRC1913 TAC MD2496
06/04/07	ML071580252	AP1000 COL Response to Requests for Additional Information (TR #39).	AP1000 DCP/NRC1918 TR #39

Document Date	Accession Number	Title	Case/Reference Number
06/07/07	ML071630144	AP1000 COL Response to Request for Additional Information (TR #33).	DCP/NRC1925
06/07/07	ML071630150	AP1000 COL Response to Request for Additional Information (TR #34).	DCP/NRC1926
06/07/07	ML071630154	AP1000 COL Response to Request for Additional Information (TR #35).	DCP/NRC1923
06/07/07	ML071630161	AP1000 COL Response to Request for Additional Information (TR #44).	DCP/NRC1924
06/08/07	ML071640053	Transmittal of AP1000 COL Response to Requests for Additional Information (TR #28).	AP1000 APP-GW-GLN-024, Rev 0 DCP/NRC1928 TR #28
06/08/07	ML071640055	Enclosure - RAI-TR28-001, Rev 0, "Response to Requests for Additional Information on Technical Report No. 28."	AP1000 APP-GW-GLR-024, Rev 0
06/08/07	ML071640071	AP1000 COL Response to Request for Additional Information (TR #59).	AP1000APP-GW-GLR-011, Rev 0DCP/NRC1927TR #59
06/08/07	ML071640219	Enclosure 3 - Response to Requests for Additional Information on Technical Report No. 31, RAI-TR3 1-001 and RAI-TR31-002.	DPC/NRC1914 RAI-TR31-001 RAI-TR31-002 TR #31
06/08/07	ML071640243	AP1000 COL Response to Request for Additional Information (TR #54).	AP1000 DCP/NRC1929 TAC MD2551 TR #54
06/14/07	ML071690098	AP1000 COL Response to Request for Additional Information (TR #54).	AP1000 DCP/NRC1938 TAC MD2551 TR #54
06/21/07	ML071730474	Westinghouse Electric Company, AP1000 COL Response to Request for Additional Information (TR 6).	APP-GW-GLR-021 DCP/NRC1947
07/05/07	ML071870409	AP1000 COL Response to Request for Additional Information (TR 3).	DCP/NRC1954 TAC MD2358
07/05/07	ML071870411	AP1000 COL Response to Requests for Additional Information (TR 57).	DCP/NRC1955
07/12/07	ML071980057	AP1000 COL Responses to Requests for Additional Information (TR #74A).	AP1000 APP-GW-GLR-064, Rev 0 DCP/NRC1961 TAC MD3838 TR #74A

Document Date	Accession Number	Title	Case/Reference Number
07/13/07	ML071980059	AP1000 COL Responses to Requests for Additional Information (TR #45).	AP1000 APP-GW-GLR-027, Rev 1 DCP/NRC1960 TAC MD2495 TR #45
07/17/07	ML072010044	Submittal of Response to NRC Request for Additional Information on AP1000 Standard Combined License Technical Report 44, APP-GW-GLR-026, Rev. 0, New Fuel Storage Rack Structural/Seismic Analysis.	DCP/NRC1963TR #44
07/17/07	ML072010045	Enclosure 2, Proprietary Response to Request for Additional Information on Technical Report No. 44.	RAI-TR44-012
07/17/07	ML072010046	AP1000 COL Responses to Requests for Additional Information (TR#93).	AP1000 APP-GW-GLR-073, Rev. 0 DCP/NRC1958 RAI-TR93-ICE2-01 RAI-TR93-ICE2-05 TAC MD4624 TR #93
07/17/07	ML072010048	AP1000 COL Responses to Requests for Additional Information (TR #44).	AP1000 APP-GW-GLR-026, Rev 0 DCP/NRC1962 TAC MD2104 TR #44
07/18/07	ML072040015	AP1000 COL Response to Requests for Additional Information (TR #24).	APP-GW-GLR-060, Rev 0 AW-07-2305 DCP/NRC1959
07/18/07	ML072040016	Response to Requests for Additional Information on Technical Report No. 24, RAI-TR24-EMB2-02 and RAI-TR24-EMB2-04.	APP-GW-GLR-060, Rev 0 AW-07-2305 DCP/NRC1959
07/27/07	ML072130066	AP1000 COL Response to Requests for Additional Information (TR 6).	DCP/NRC1970
07/27/07	ML072130068	AP1000 COL Responses to Requests for Additional Information (TR #86).	AP1000 DCP/NRC1969 RAI-TR86-SBPB-01, Rev 0 TR #86
07/27/07	ML072130070	AP1000 COL Response to Requests for Additional Information (TR #59).	AP1000 DCP/NRC1966 RAI-TR59-COLP-011, Rev 0 TR #59
08/21/07	ML072350135	Westinghouse Electric Company, AP1000 COL Response to Request for Additional Information (TR #12).	DCP/NRC1978

Document Date	Accession Number	Title	Case/Reference Number
08/21/07	ML072350140	AP1000 COL Responses to Requests for Additional Information (TR #24).	AP1000 DCP/NRC1977
08/21/07	ML072350220	AP1000 COL Responses to Requests for Additional Information (TR #100).	AP1000 DCP/NRC1976 TR #100
08/21/07	ML072350225	Westinghouse Electric Company - AP1000 COL Responses to Requests for Additional Information (TR #52).	AP1000 DCP/NRC 1975 TR #52
08/21/07	ML072350233	AP1000 COL Response to Request for Additional Information (TR #10).	AP1000 DCP/NRC1972
08/23/07	ML072390022	AP1000 COL Responses to Requests for Additional Information (TR #93).	AP1000 DCP/NRC1982 RAI-TR93-ICE2-06 RAI-TR93-ICE2-07 TR #93
08/23/07	ML072390023	AP1000 COL Response to Requests for Additional Information (TR 102).	AP1000 DCP/NRC1981 RAI-TR102-SPLA-01 RAI-TR102-SPLA-08 TR #102
08/23/07	ML072390024	AP1000 COL Response to Requests for Additional Information (TR 3).	AP1000 APP-GW-S2R-010 DCP/NRC-1858 DCP/NRC1942 DCP/NRC1980 RAI-TR03-021 RAI-TR03-023 TR #03
08/31/07	ML072470706	AP1000 COL Responses to Requests for Additional Information (TR #11e).	DCP/NRC1984
08/31/07	ML072470708	AP1000 COL Responses to Requests for Additional Information (TR #16).	DCP/NRC1985
08/31/07	ML072480187	AP1000 COL Response to Requests for Additional Information (TR 3).	DCP/NRC1987TAC MD2358
09/05/07	ML072500015	AP1000 COL Response to Requests for Additional Information (TR 9).	DCP/NRC1986
09/07/07	ML072530888	AP1000 COL Responses to Requests for Additional Information (TR #94).	APP-GW-GLR-066, Rev 0 DCP/NRC1989
09/07/07	ML072530977	AP1000 COL Responses to Requests for Additional Information (TR #35).	APP-GW-GLN-010, Rev 0 DCP/NRC1991

Document Date	Accession Number	Title	Case/Reference Number
09/07/07	ML072530978	AP1000 COL Responses to Requests for Additional Information (TR #97).	APP-GW-GLN-022, Rev. 1 DCP/NRC1992 RAI-TR97-ICE-01 RAI-TR97-ICE-02 RAI-TR97-ICE-03
09/18/07	ML072620349	AP1000 COL Response to Requests for Additional Information (TR 107).	AP1000 DCP/NRC1998 TR 107
09/18/07	ML072620351	AP1000 COL Response to Requests for Additional Information (TR 85).	DCP/NRC1999
09/19/07	ML072640085	AP1000 COL Response to Requests for Additional Information (TR 106).	AP1000 DCP/NRC2000 TR 106
09/21/07	ML072670544	AP1000 COL Response to Requests for Additional Information (TR 85).	DCP/NRC2002
09/28/07	ML072750136	AP1000 COL Response to Requests for Additional Information (TR 85).	AP1000 APP-GW-GLR-044 DCP/NRC2006 TR 85
09/28/07	ML072750139	AP1000 COL Response to Requests for Additional Information (TR 45).	DCP/NRC2007
09/28/07	ML072750140	AP1000 COL Responses to Requests for Additional Information (TR 18).	DCP/NRC2010
09/28/07	ML072750143	AP1000 COL Response to Request for Additional Information (TR 34).	DCP/NRC2008
10/04/07	ML072830018	AP1000 COL Response to Requests for Additional Information (TR 121).	AP1000DCP/NRC2015RAI-TR 121-CHPB-01, Rev 0TR 121
10/04/07	ML072830050	AP1000 COL Response to Request for Additional Information (TR 36).	AP1000 DCP/NRC2017 RAI-TR36-012, Rev 1 TR 36
10/05/07	ML072830016	AP1000 COL Response to Requests for Additional Information (TR 106).	AP1000 DCP/NRC2019 RAI-TR106-CIB1-01, Rev 0 TR 106
10/05/07	ML072830017	AP1000 COL Response to Requests for Additional Information (TR 34).	DCP/NRC2018
11/16/07	ML073240107	AP1000 COL Responses to Requests for Additional Information (TR 52).	DCP/NRC2042
12/04/07	ML073410094	AP1000 COL Responses to Requests for Additional Information (TR 85).	DCP/NRC2050
12/07/07	ML073450093	AP1000 COL Response to Request for Additional Information (TR 66).	AP1000 DCP/NRC2053 TR 66

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12/07/07	ML073450095	AP1000 COL Responses to Requests for Additional Information (TR 79).	DCP/NRC2052
12/19/07	ML073551153	AP1000 COL Response to Requests for Additional Information (TR 98).	APP-GW-GLN-098 DCP/NRC2062 TR 98
06/20/08	ML081760189	AP1000 Response to Requests for Additional Information (SRP9.1.2).	DCP/NRC2167
06/20/08	ML081760190	Westinghouse Electric Co. - AP1000 Response to Requests for Additional Information (TR 54).	DCP/NRC2166
06/20/08	ML081760191	AP1000 Response to Requests for Additional Information (SRP10.4.7).	AP1000 DCP/NRC2165 SRP10.4.7
06/20/08	ML081760192	AP1000 Response to Requests for Additional Information (SRP3.9.5).	AP1000DCP/NRC2164
06/20/08	ML081760193	AP1000 Response to Requests for Additional Information (SRP3.9.2).	AP1000 DCP/NRC2163
06/20/08	ML081760195	AP1000 Response to Requests for Additional Information (SRP14.2).	AP1000 DCP/NRC2162
06/20/08	ML081760196	AP1000 Response to Requests for Additional Information (SRP9.3.3).	AP1000 DCP/NRC2169
06/20/08	ML081780174	AP1000 Response to Requests for Additional Information (SRP3.12).	DCP/NRC2173
06/20/08	ML081780175	AP1000 Response to Requests for Additional Information (SRP10).	DCP/NRC2174
06/20/08	ML081780176	AP1000 Response to Requests for Additional Information (SRP3.6).	AP1000 DCP/NRC2171 SRP3.6
06/20/08	ML081780177	AP1000 Response to Requests for Additional Information (SRP3.6.3).	DCP/NRC2172
06/26/08	ML081820722	AP1000 Response to Requests for Additional Information (SRP2.3.1).	DCP/NRC2170
06/26/08	ML081820723	Westinghouse Electric Company, AP1000 Response to Requests for Additional Information (SRP3.9.3).	DCP/NRC2176
06/26/08	ML081820724	AP1000 Response to Requests for Additional Information (SRP9.1.4).	DCP/NRC2177

Document Date	Accession Number	Title	Case/Reference Number
06/26/08	ML081820725	Westinghouse Electric Company - AP1000 Response to Requests for Additional Information (SRP9.1.5).	DCP/NRC2178
06/26/08	ML081820726	AP1000 Response to Requests for Additional Information (SRP9.2.1).	DCP/NRC2179
06/26/08	ML081820727	AP1000 Response to Requests for Additional Information (SRP9.2.2).	DCP/NRC2180
06/30/08	ML081850613	Westinghouse Electric Company - AP1000 Response to Requests for Additional Information (SRP 15.3).	DCP/NRC2187
07/01/08	ML081900155	AP1000 Response to Request for Additional Information (SRP3.8.4).	AP1000 DCP/NRC2189
07/03/08	ML081900156	AP1000 Response to Request for Additional Information (SRP5.4.1).	AP1000DCP/NRC2191
07/03/08	ML081900157	AP1000 Response to Request for Additional Information (SRP3.6.1).	AP1000 DCP/NRC2192
07/03/08	ML081900158	AP1000 Response to Request for Additional Information (SRP4).	DCP/NRC2193
07/03/08	ML081900159	AP1000 Response to Request for Additional Information (SRP3.4.1).	DCP/NRC2194
07/07/08	ML081910138	AP1000 Response to Request for Additional Information (SRP7).	AP1000 DCP/NRC2195 SRP7
07/08/08	ML081920166	AP1000 Response to Request for Additional Information (TR96), Submitted in Support of the AP1000 Design Certification Amendment Application.	AP1000 APP-GW-GLR-067 DCP/NRC2196 TR 96
07/08/08	ML081920167	Westinghouse Response to Request for Additional Information on AP1000 Standard Combined License Technical Report (TR) 49, APP-GW-GLR-062, AP1000, "Security Enhancement Report."	APP-GW-GLR-062 DCP/NRC2197
07/08/08	ML081920168	Response to Request for Additional Information on AP1000 Technical Report Review.	DCP/NRC2197 RAI-TR49-NSIR-01
07/11/08	ML081970176	AP1000 Response to Request for Additional Information (SRP14.2), in Support of Design Certification Amendment Application.	AP1000 DCP/NRC2199 SRP14.2

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07/11/08	ML081970177	AP1000 Response to Request for Additional Information (SRP8.3.1), in Support of Design Certification Application.	DCP/NRC2198
07/14/08	ML081980185	AP1000 Response to Request for Additional Information (RAI) on SRP Section 18, Submitted in Support of Design Certification Amendment Application, Completing Six of Fifteen Requests Received to Date.	DCP/NRC2201
07/14/08	ML081980186	Westinghouse Electric Co., AP1000 Response to Request for Additional Information on SRP Section 3.9.6.	AP1000 DCP/NRC2200
07/15/08	ML081990375	AP1000 Response to Requests for Additional Information (TR 44).	DCP/NRC2203 TR 44
07/17/08	ML082040273	AP1000 Response to Request for Additional Information, Applying to All COL Applications Referencing AP1000 Design Certification and AP1000 Design Certification Amendment Application.	DCP/NRC2204
07/17/08	ML082040274	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 5.4 in Support of AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2205
07/18/08	ML082040275	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP5.2.1).	DCP/NRC2206
07/18/08	ML082040276	AP1000 Response to Request for Additional Information (SRP3.9.6) in Support of the Design Certification Amendment Application.	DCP/NRC2207
07/18/08	ML082040277	AP1000 Response to Request for Additional Information (SRP13.3).	DCP/NRC2210
07/18/08	ML082040301	Westinghouse, Response to Request for Additional Information on SRP Section 10.3 in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2208

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07/18/08	ML082040302	AP1000 Response to Request for Additional Information (SRP6) in Support of Design Certification Amendment Application.	DCP/NRC2209
07/22/08	ML082060194	Westinghouse Electric Co., Submittal of Response to Request for Additional Information on SRP Section 19.0.	AP1000 DCP/NRC2211
07/22/08	ML082060195	Westinghouse, Submittal of Response to Request for Additional Information on SRP Section 3.8.2.	AP1000 DCP/NRC2212
07/24/08	ML082100163	AP1000 Response to Request for Additional Information (SRP10.4.7) in Support of the Design Certification Amendment Application.	DCP/NRC2214
07/24/08	ML082100164	AP1000 Response to Request for Additional Information (SRP3.9.6).	AP1000 DCP/NRC2213 SRP3.9.6
07/29/08	ML082140228	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 15 in Support of AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2216
07/29/08	ML082170396	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 13 in Support of AP1000 Design Certification Amendment Application.	AP1000DCP/NRC2215
07/31/08	ML082180121	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 9.5.2 in Support of AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2220
07/31/08	ML082180122	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 10.2 in Support of AP1000 Design Certification Amendment Application.	CP/NRC2219
08/04/08	ML082190361	AP1000 Response to Request for Additional Information on Technical Report (TR) 66, APP-GW-GLR-070, "Development of Severe Accident Management Guidance."	DCP/NRC2218

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08/04/08	ML082200546	Westinghouse Electric Company, AP1000 Response to Request for Additional Information (SRP 18).	DCP/NRC2217
08/04/08	ML082200547	AP1000 Response to Request for Additional Information (SRP4), in Support of the Design Certification Amendment Application.	DCP/NRC2221
08/15/08	ML082330097	AP1000 Response to Request for Additional Information (SRP5.2.2).	AP1000 DCP/NRC2224 SRP5.2.2
08/15/08	ML082330098	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP 16.1.1) for RAI-SRP16.1.1-SEB1-01, in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2229 SRP16.1.1
08/15/08	ML082330099	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP8.3) for for RAI-SRP8.3-EEB-01 through -05, in Support of the AP1000 Design Certification Amendment Application.	DCP/NRC2223
08/20/08	ML082350324	Westinghouse, AP1000 Response to Request for Additional Information on APP-GW-S2R-010, Technical Report Number 3 (TR03).	APP-GW-S2R-010 DCP/NRC2225 TR03
08/20/08	ML082350325	AP1000 Response to Request for Additional Information (RAI), (SRP11.3).	AP1000DCP/NRC2231SRP11.3
08/20/08	ML082350326	AP1000 Response to Request for Additional Information (RAI) (SRP2.5).	AP1000 DCP/NRC2230 SRP2.5
08/21/08	ML082390115	AP1000 Response to Requests for Additional Information (SRP9.1.2), in Support of the Design Certification Amendment Application.	DCP/NRC2234
08/21/08	ML082390116	AP1000 Response to Request for Additional Information (SRP3.10) in Support of Design Certification Amendment Application.	DCP/NRC2235

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08/21/08	ML082390117	AP1000 Response to Request for Additional Information (SRP19.0) in Support of the Design Certification Amendment Application.	DCP/NRC2233
08/22/08	ML082390029	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (TR 94), in Support of the AP1000 Design Certification Amendment Application.	AP1000 APP-GW-GLR-066, Rev 1 DCP/NRC2237
08/22/08	ML082390112	Westinghouse Electric Co., Withdrawal of AP1000 Response to Request for Additional Information, SRP8.3.1-EEB-03.	AP1000 DCP/NRC2236
08/28/08	ML082460513	Westinghouse Submittal of a Revised Response to the NRC Request for Additional Information (RAI) on SRP Section 3.9.2 in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2241 SRP3.9.2
08/28/08	ML082460517	Westinghouse Submittal a Revised Response to the NRC Request for Additional Information (RAI) on SRP Section 3.9.5 in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2239 SRP3.9.5
08/29/08	ML082470022	Westinghouse Electric Co., Response for Additional Information on TR 94 (APP-GW-GLR-066 Rev. 1) in Support of the AP1000 Design Certification Amendment Application.	APP-GW-GLR-066, Rev 1 DCP/NRC2238
08/29/08	ML082480501	AP1000 Response to Request for Additional Information (SRP3.12) in Support of the Design Certification Amendment Application.	DCP/NRC2240
08/29/08	ML082480520	AP1000 Response to Request for Additional Information (SRP14.2), in Support of the Design Certification Amendment Application.	DCP/NRC2242
09/03/08	ML082520228	AP1000 Response to Request for Additional Information (SRP9.1.5) in Support of the Design Certification Amendment Application.	DCP/NRC2246

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09/03/08	ML082520229	AP1000 Response to Request for Additional Information (SRP7) In Support of the Design Certification Amendment Application.	DCP/NRC2245
09/05/08	ML082520821	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP19.0), for RAI-SRP19.0-SPLA-05, in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2247
09/05/08	ML082520823	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP 17.4) for RAI-SRP17.4-SPLA-03, in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2248
09/05/08	ML082520826	AP1000 Response to Request for Additional Information (SRP5.3.2) - for RAI-SRP5.3.2-CIB1-01 and -02.	DCP/NRC2250
09/05/08	ML082520828	Westinghouse Electric Company, AP1000 Response to Request for Additional Information (SRP3.7.2), for RAI-SRP3.7.2-SEB1-03, in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2249
09/05/08	ML082520830	Westinghouse Electric Company, AP1000 Response to Request for Additional Information (SRP3.2), in Support of the AP1000 Design Certification Amendment Application for RAI-SRP3.2.1-EMB2-01,-02, and -03 and RAI-SRP3.2.2-EMB2-01,-02, and -03.	AP1000 DCP/NRC2251
09/05/08	ML082520832	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP3.9.1) for RAI-SRP3.9.1-EMB1-01 and -02, in Support of the AP1000 Design Certification Amendment Application.	AP1000DCP/NRC2252

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09/05/08	ML082530040	Westinghouse, Response to the NRC Request for Additional Information (RAI) on SRP Section 3.8.4 in Support of the AP1000 Design Certification Amendment Application.	DCP/NRC2253
09/09/08	ML082560236	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP14.3).	DCP/NRC2259
09/09/08	ML082560237	Westinghouse Electric Company - AP1000 Response to Request for Additional Information (SRP 12) for RAI-SRP12.1-CHPB-01 and RAI-SRP12.2-CHPB-01 and -02, in support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2257 SRP12
09/09/08	ML082560238	AP1000 Response to Request for Additional Information (SRP1 1.2).	DCP/NRC2256 SRP11.2
09/09/08	ML082560239	AP1000 Response to Request for Additional Information (SRP3.9.6) In Support of the Design Certification Amendment Application.	DCP/NRC2255
09/09/08	ML082560240	AP1000 Response to Request for Additional Information (SRP2.5).	AP1000 DCP/NRC2254 SRP2.5
09/09/08	ML082560241	Westinghouse Electric Company - Transmittal of AP1000 Response to Request for Additional Information (SRP7.1) for RAI-SRP7.1-ICE-15 in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP/NRC2258 SRP7.1
09/09/08	ML082560242	Enclosure 3 - Response to Request for Additional Information on SRP Section 7.1 RAI-SRP7.1-ICE-15.	AP1000 DCP/NRC2258
05/14/09	ML091390055	AP1000 Response to Request for Additional Information (SRP 6) in Support of the Design Certification Amendment Application.	DCP/NRC2477
05/27/09	ML091520091	AP1000 Response to Request for Additional Information (SRP 6) in Support of the Design Certification Amendment Application.	DCP/NRC2502

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05/27/09	ML091540301	AP1000 Security Related Response to Request for Additional Information (SRP 6) in Support of the Design Certification Amendment Application.	DCP/NRC2502
11/09/09	ML093170459	Enclosure 3, Response to Request for Additional Information on SRP Section 7.1, RAI-SRP7.1-FMEA-02.	DCP_NRC_002676
11/09/09	ML093420275	Westinghouse - AP1000 Response to Request for Additional Information (TR 44).	AP1000 DCP_NRC_002688 TR 44
11/11/09	ML093200643	AP1000 Response to Request for Additional Information (SRP 9), in Support of the Design Certification Amendment Application.	DCP_NRC_002690
12/28/09	ML100050275	AP1000 Response to Request for Additional Information (SRP 10), RAI-SRP10.2-SBPA-02 R3, in Support of the Design Certification Amendment Application.	DCP_NRC_002723
02/11/10	ML100480094	AP1000 Response to Request for Additional Information (SRP 16), RAI-SRP16-CTSB-20 R1, in Support of the Design Certification Amendment Application.	DCP_NRC_002773
02/11/10	ML100480096	AP1000 Response to Request for Additional Information (SRP 7) RAI-SRP7.8-DAS-01, RAI-SPR7.8-DAS-3 and RAI-SRP7.8-DAS-04 in Support of the Design Certification Amendment Application.	DCP_NRC_002772
04/07/10	ML101020695	AP1000 Response to Request for Additional Information RAI-SRP5.2.3-CIB1-01 R2.	DCP_NRC_002847
04/12/10	ML101040256	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 2 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002849
04/16/10	ML101090299	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 6 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002851

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04/20/10	ML101121005	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 17 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002853
04/21/10	ML101121004	AP1000 Response to Request for Additional Information (SRP 5).	AP1000 DCP_NRC_002854 SRP 5
04/26/10	ML101180082	Westinghouse Electric Company, Response to Request for Additional Information on SRP Section 6.2.2 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002859
04/26/10	ML101180083	Response to Request for Additional Information on SRP Section 6.2.2 RAI SRP6.2.2-CIB1-28 R1.	DCP_NRC _002859
04/26/10	ML101230335	AP1000 Response to Request for Additional Information (SRP6.2.2), RAI SRP6.2.2-CIB1-31 in Support of the Design Certification Amendment Application.	AW-10-2806 DCP_NRC_002860
04/26/10	ML101230336	Enclosure 3 to DCP_NRC_002860 - Response to Request for Additional Information on SRP Section 6.2.2 RAI SRP6.2.2-CIB1-31.	DCP_NRC_002860
04/30/10	ML101230613	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 2 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002861
04/30/10	ML101241173	AP1000 Response to Request for Additional Information (SRP 3), RAI-SRP3.9.4-EMB1-01 in Support of the Design Certification Amendment Application.	DCP_NRC_002862
05/06/10	ML101300517	Westinghouse Electric Company - Response to Request for Additional Information (SRP 16), RAI-SRP16.3-CTSB-SCP-1 R1, in Support of the AP1000 Design Certification Amendment Application.	AP1000 DCP_NRC_002864 SRP 16

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05/07/10	ML101340549	AP1000 Response to Proposed Open Item (Chapter 8), OI-SRP8.3.2-EEB-09 R2 in Support of the Design Certification Amendment Application.	DCP_NRC_002866
05/10/10	ML101320314	2010/05/10 AP-1000 DCD Review - OI-SRP9.1.3-SBPA-13 SFP AP1000 (1A)] for Review / Comments Due: 5/13/2010	
05/10/10	ML101320318	2010/05/10 AP-1000 DCD Review - OI-SRP9.1.5-SBPB-01 AP1000 (2A)] for Review / Comments Due: 5/13/2010	
05/10/10	ML101340545	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section 8 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002868
05/11/10	ML101330142	AP1000 Response to Request for Additional Information (RAI-TR94-NSIR-20, Revision 1).	AP1000 DCP_NRC_002871
05/11/10	ML101340456	AP1000 Response to Request for Additional Information (SRP 8), OI-SRP8.3.2-EBB-08 R1 in Support of the Design Certification Amendment Application.	DCP_NRC_002870
05/11/10	ML101340590	AP1000 Response to Request for Additional Information (SRP6.2.2), RAI SPR6.2.2-SPCV-25 R1, in Support of the Design Certification Amendment Application.	AW-10-2816 DCP_NRC_002872
05/11/10	ML101340591	Enclosure 3 to DCP_NRC_002872, Response to Request for Additional information on SRP Section 6.2.2, RAI-SRP6.2.2-SPCV-25 R1.	DCP_NRC_002872
05/12/10	ML101340551	AP1000 Response to Request for Additional Information (TR 85), RAI-TR85-SEB1-10, R4 in Support of the Design Certification Amendment Application.	DCP_NRC_002873

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05/13/10	ML101380126	Westinghouse Electric Co. - Response to Request for Additional Information on SRP Section 6 in Support of AP1000 Design Certification Amendment Application.	AP1000 DCP_NRC_002875 SRP 6
05/14/10	ML101380123	AP1000 Response to Request for Additional Information (TR 85) RA1-TR85-SEB1-37 R4 in Support of the Design Certification Amendment Application.	DCP_NRC_002878
05/14/10	ML101380124	AP1000 Response to Request for Additional Information (SRP 3), RAI-SRP3.7.1-SEB1-18, in Support of the Design Certification Amendment Application.	DCP_NRC_002877
05/14/10	ML101380127	Response to Request for Additional Information on SRP Section 11 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002876
05/17/10	ML101390225	Westinghouse Electric Company, Response to Request for Additional Information (SRP 6), RAI-SRP6.2.4-SPCV-04, in support of the AP1000 Design Certification Amendment Application.	DCP_NRC_002880
05/19/10	ML101440391	AP1000 Response to Proposed Open Item (Chapter 9) OI-SRP9.1.3-SBPA-13 R1, OI-SRP9.1.4-SBPA-03 R2 & OI-SRP9.1.5-SBPB-01 R2.	DCP_NRC_002881
05/21/10	ML101450248	Westinghouse Electric Company, AP 1000 Response to Request for Additional Information (SRP 18), in support of the AP1000 Design Certification Amendment Application.	DCP_NRC_002883
05/21/10	ML101460209	Westinghouse Electric Company, Response to Request for Additional Information (SRP 3), RAI-SRP3.9.4-EMB1-01 R1, in Support of the AP1000 Design Certification Amendment Application.	DCP_NRC_002888

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05/24/10	ML101470410	AP1000 Response to Request for Additional Information (SRP 6), RAI-SRP6.4-SPCV-15 R1 in Support of the Design Certification Amendment Application, Cover Letter through AP1000 Technical Report Review Page 46-134.	DCP_NRC_002885
05/24/10	ML101470411	AP1000 Response to Request for Additional Information (SRP 6), RAI-SRP6.4-SPCV-15 R1 in Support of the Design Certification Amendment Application, AP1000 Technical Report Review Page 47-134 through End.	DCP_NRC_002885
06/14/10	ML101670133	AP1000 Response to Request for Additional Information (SRP6.2.2), RAI-SRP6.2.2-SRSB-44 in Support of the Design Certification Amendment Application.	AW-10-2853 DCP_NRC_002917
07/09/10	ML101960267	AP1000 Response to Request for Additional Information (SRP 3) in Support of the Design Certification Amendment Application, Cover Letter.	DCP_NRC_002952
07/13/10	ML101970025	AP1000 Response to Request for Additional Information (TR54) RAI-TR54-01 R1.	DCP_NRC_002963
07/13/10	ML101970026	AP1000 Response to Request for Additional Information RAI-TR03-037.	AP1000 DCP_NRC_002964
07/15/10	ML102000079	AP1000 Response to Request for Additional Information (SRP 3).	AP1000 DCP_NRC_002966 SRP 3
07/15/10	ML102000080	AP1000 Response to Request for Additional Information (SRP TR85).	AP1000 DCP_NRC_002965 SRP TR85
07/20/10	ML102040034	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section TR54 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002976
07/20/10	ML102040035	Westinghouse Electric Co., Response to Request for Additional Information on SRP Section TR44 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_002975

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09/13/10	ML102630020	Westinghouse, AP1000 Response to Request for Additional Information (TR44).	DCP_NRC_003044 TR44
09/15/10	ML102630017	AP1000 Response to Request for Additional Information (SRP 19).	AP1000 DCP_NRC_003045 SRP 19
09/15/10	ML102630018	AP1000 Technical Report Review, Enclosure 1 to DCP_NRC_003045.	AP1000 DCP_NRC_003045 SRP 19
09/23/10	ML102670584	AP1000 Response to Request for Additional Information (TR85).	AP1000 DCP_NRC_003049 TR85
09/29/10	ML102770328	AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-005, Revision 4 (TR-09).	DCP_NRC_003051 TR-09
09/29/10	ML102770329	Westinghouse Revised RAI Response to RAI-SRP3.9.1-EMB-03, Rev. 2.	DCP_NRC_003046
09/30/10	ML102780277	Enclosure 1 to DCP_NRC_003052, AP1000 Response to Request for Additional Information (SRP 19), RAI-SRP19F-AIA-01 R2, and RAI-SRP19F-AIA-09 R1.	DCP_NRC_003052
09/30/10	ML102780286	AP1000 Response to Request for Additional Information (SRP 19), RAI-SRP19F-AIA-01 R2, and RAI-SRP19F-AIA-09 R1.	DCP_NRC_003052
10/19/10	ML102940228	Enclosure 3 - Westinghouse Markup of NRC Chapter 6 ASE Review Indicating Proprietary Sections.	DCP_NRC_003067
10/21/10	ML102990048	Westinghouse, Response to Request for Additional Information on SRP Section 3 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_003071
10/21/10	ML102990054	AP1000 Response to Request for Additional Information on SRP Section 3.	DCP_NRC_003070
10/21/10	ML102990055	AP1000 Response to Request for Additional Information (SRP 3) (Proprietary).	DCPNRC_003070
10/22/10	ML103010046	Westinghouse Electric Company, Response to Request for Additional Information (TR85) in support of the AP1000 Design Certification Amendment Application.	

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11/12/10	ML103210409	Reply to Notice of Violation Cited in NRC Inspection Report No. 05200006/2010-203 dated October 28, 2010.	DCP_NRC_003084 IR 10-203
11/12/10	ML103210455	Westinghouse - Response to Request for Additional Information on SRP Section 10 in Support of AP1000 Design Certification Amendment Application.	DCP_NRC_003088 SRP 10
11/19/10	ML103470566	AP1000 Response to Request for Open Item (SRP 19), RAI-SRP19F-AIA-01 R3 and RAI-SRP19F-AIA-09 R2 In Support of the Design Certification Amendment Application.	DCP_NRC_003093
11/19/10	ML103470567	Enclosure 1 to DCP_NRC_003093, AP1000 Response to Request for Open Item (SRP 19), RAI-SRP19F-AIA-01 R3 and RAI-SRP19F-AIA-09 R2 In Support of the Design Certification Amendment Application.	DCP_NRC_003093
11/22/10	ML103300210	AP1000 Response to Request for Additional Information (SRP 19).	DCP_NRC_003094 RAI-SRP19F-AIA-01, Rev 4
11/22/10	ML103300211	Enclosure 1, Response to Request for Additional Information on SRP Section 19.	DCP_NRC_003094 RAI-SRP19F-AIA-01, Rev 4 SRP 19
1/28/2011	ML110330046	AP1000 Response to Request for Additional Information (CI-SRP9.1.1 -SRSB-01 and RAI-SRP4.3-SRSB-03).	AW-11-3067 CI-SRP9.1.1-SRSB-01 DCP_NRC_003114 RAI-SRP4.3-SRSB-03
2/9/2011	ML110450101	AP1000 Response to Request for Additional Information (SRP 3).	AP1000 DCP_NRC_003124 SRP 3
3/1/2011	ML110630109	Response to Requests for Additional Information on PCS Operation.	DCP_NRC_003147

F. REPORTS BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

1. Report on the Final Safety Evaluation Report Associated with the Amendment to the Ap1000 Design Control Document, December 13, 2010

2. Long-Term Core Cooling for the Westinghouse AP1000 Pressurized Water Reactor, December 20, 2010

3. Report on the Safety Aspects of the Aircraft Impact Assessment tor the Westinghouse AP1000 Design Certification Amendment Application, January 19, 2011

4. Response to the February 5, 2011, EDO Letter Regarding the Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document," May 19, 2011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 13, 2010

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

Dear Chairman Jaczko:

During the 578th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 2-4, 2010, we reviewed the NRC staff's Advanced Final Safety Evaluation Report (AFSER) for the pending AP1000 Design Certification Amendment (DCA) application. The amendment is to be reflected in a revision to the AP1000 Design Control Document (DCD). The amendment involves changes to Tier 1 information, and its approval will require rulemaking. We had a number of subcommittee and full committee meetings to review the technical aspects of the amendment. During these meetings, we had the benefit of discussions with representatives of the NRC staff, Westinghouse (WEC), and members of the public. We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

The changes proposed in the AP1000 DCA maintain the robustness of the previously certified design. We conclude that there is reasonable assurance that the revised design can be built and operated without undue risk to the health and safety of the public. This conclusion is contingent on the results of our concurrent reviews of the aircraft impact assessment and long-term core cooling issues which will be discussed in separate letters.

This conclusion relies in part on information and commitments provided by WEC during the course of our meetings which have not yet been confirmed to be included in the DCA application. This information and commitments are noted in the discussion following, and the staff should ensure they are appropriately documented as part of the DCA.

BACKGROUND

For its initial design approval and certification of the AP1000 design, the NRC issued NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Design," in September 2004 and published the proposed design certification rule on April 18, 2005. In December 2005, the NRC staff evaluated the conforming Revision 15 to the AP1000 DCD in Supplement 1 to NUREG-1793. The NRC published a final rule certifying the AP1000 standard plant design on January 27, 2006.

Thus, the existing AP1000 certification rule is reflected in DCD Revision 15. Revision 18 was submitted by WEC in a letter dated December 1, 2010, and it includes changes identified in Revision 16, submitted May 26, 2007, and in Revision 17, submitted September 22, 2008, as well as those changes made subsequent to submittal of Revision 17 which are identified in the AFSER, Chapter 23.

In addition, WEC submitted letters to supplement its DCA application dated October 26, November 2, and December 12, 2007, as well as January 11, and 14, 2008. Finally, NuStart Energy Development, LLC and WEC submitted a number of technical reports (TRs) for review. TRs typically address a topical area, such as the design of a component, structure, or process, in support of the AP1000 design.

The DCA application proposes to incorporate changes in the AP1000 certification rule reflecting the following:

- Design standardization, which was enhanced by elimination of numerous combined license (COL) open items currently in the existing rule.
- New regulatory requirements, including requirements related to aircraft impact. (As previously noted, review of compliance with the aircraft impact requirements will be discussed in a separate letter).
- Design finalization, which was required to produce construction drawings and procurement specifications. This includes reduced reliance on design acceptance criteria (DAC).

Significant changes proposed in the DCA application include the following:

- Redesign of the shield building to use a modular, steel concrete composite (SC) structure, replacing the existing reinforced concrete (RC) design. The redesign reduces passive heat removal air flow and affects seismic, aircraft impact, and other loading analyses.
- Redesign of the Reactor Vessel Support System to increase stiffness.
- Increase in the range of foundation soil conditions considered.
- Closure of four digital instrumentation and control (DI&C) DAC, with only one remaining open. Numerous I&C changes were made to reflect design evolution, such as addition of a reactor trip function, implementation of a rod withdrawal prohibit, and modification of the containment isolation logic for the Component Cooling System.
- Closure of four human factors engineering (HFE) DAC, with none remaining open.
- Modification of the reactor coolant pump (RCP) design, including an increase in its rotational inertia.
- Addition of a flow skirt at the inlet to the reactor vessel lower plenum.
- Redesign of the Steam and Power Conversion Systems.

Our review of the DCA application began with a status review by the Full Committee during the 562nd meeting in May 2009. Subsequently, our AP1000 subcommittee held 12 meetings, totaling 21 days of meetings, as listed in the appendix to this letter.

DISCUSSION

Shield Building Redesign

The AP1000 shield building described in AP1000 DCD, Revision 15, is an RC design. In AP1000 DCD, Revisions 16 and 17, WEC proposed a new shield building design. The new design includes provisions to meet the requirements of the new aircraft impact rule, 10 CFR 50.150. (As indicated previously, the results of our review for compliance with the aircraft impact rule will be reported in a separate letter).

The key features of the new shield building are: a cylindrical wall which comprises the bulk of the structure constructed of SC modules; a conical RC roof structure with an integral RC water tank which contains approximately 7 million pounds of water; a tension ring at the intersection of the roof with the cylindrical wall consisting of a built-up closed section of steel plates filled with concrete; and mechanical connections that join the SC wall to the basemat and the RC wall of the auxiliary building.

The tension ring is designed as a steel structure in accordance with the American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690. The steel frame for the roof is designed to the applicable building code, ANSI/AISC N690. The concrete roof is designed to American Concrete Institute (ACI) 349 requirements without credit for the steel plate on the bottom of the concrete. The SC modules have not been used previously in nuclear construction in the United States and were a focus of our review.

In the initial design proposed for the new shield building, the SC wall module for the 3-foot thick cylindrical wall consisted of steel faceplates with attached 6-inch long steel studs which are embedded in the 35-inch thick concrete fill between the two plates. In a letter dated October 15, 2009, the NRC staff determined that this design would require modifications to ensure its ability to perform its safety function under design basis loading conditions. Some key issues identified in the letter are listed below:

- The need to demonstrate the adequacy of the design and detailing of the SC module to function as a fully composite unit, as assumed in the WEC design and analysis.
- The need to demonstrate the adequacy of the design and detailing of the connection between the SC module wall and RC wall of the auxiliary building to withstand all design basis loads.
- The need to support the design and analysis of the shield building tension ring (i.e., ring girder) and the air-inlet region with a validated analysis method (i.e., benchmarked to experimental data) or by confirmatory model tests.

Staff concerns focused particularly on the lack of transverse reinforcement that would tie one faceplate to the opposite faceplate to ensure that the SC modules would function as a unit for either out-of-plane demands or in-plane demands.

WEC developed a revised design for the shield building that added tie bars welded to opposite faceplates in the SC wall modules, and also revised the design of the ring girder and the connections between the SC wall module and the RC wall. The revised SC wall module has thicker faceplates, as well as tie bars between the plates to help ensure that the module acts as a composite unit with increased out-of-plane shear strength. The spacing between the tie bars is greater in regions of the wall away from discontinuities and connections, which have low out-of-plane demands, than it is in the regions near discontinuities and the SC to RC connections, where out-of-plane shear demands are higher.

Although design codes for SC modular construction for some applications have been developed in Japan, codes and standards for the design of SC structural components do not exist in the United States. WEC used ACI-349, a design code for RC in nuclear safety-related structures, to guide their design of the SC cylindrical wall modules. Even though the scope of ACI-349 does not include SC construction, the underlying design philosophy, elastic behavior and strength for design basis loads and resilience through ductility for beyond design-basis loads, does apply. Also, the underlying assumptions on composite behavior of steel and concrete materials in RC structural elements do apply to SC structural elements.

To validate this adaptation of ACI-349, WEC conducted a testing program at Purdue University. The tests were intended (1) to demonstrate that the adaptations of ACI-349 proposed by WEC could be used to predict the out-of-plane shear strength, flexural capacity, and in-plane shear strength of SC structures and (2) to investigate the failure behavior of the SC modules.

The test results were also used to benchmark the finite element analyses performed to support the design of the shield building. WEC's approach to developing the design basis involved three levels of analysis with increasing levels of model refinement. Level 1 was used for determining the load magnitudes (seismic demands) imposed on the structure. It was a linear elastic analysis with a fairly coarse mesh that uses simplified models to account for concrete cracking. Level 2 was also a linear elastic analysis with a more refined mesh used for determining the member forces and deformation demands. Level 3 was a nonlinear analysis used to assess the region with high stresses, strains, and displacements in the shield building, such as the connection regions. Detailed submodels were used which included elements such as concrete, steel plates, studs, and tie bars. A strain-based failure criterion was selected to define acceptable limits under design-basis loads. The analysis models were benchmarked against the Purdue tests.

The Office of New Reactors (NRO) requested that the Office of Nuclear Regulatory Research (RES) provide assistance in evaluating the structural analysis, design, construction, and inspection methods for the AP1000. The findings in the RES report were used to inform the evaluation of the shield building design by the staff of NRO. RES engaged outside recognized experts in the field of reinforced concrete structures and composite structures. RES staff assessed and consolidated the inputs from each expert and performed their own independent assessment to develop their report.

The RES staff concluded that the agreement between the experimental results and the predictions of the Level 3 finite element models were adequate to benchmark the models for loads up to and beyond the design-basis safe shutdown earthquake (SSE). The RES staff also concluded that the models would provide useful predictions of SC module behavior for load levels beyond the design-basis level and below the self-imposed analysis strain limits.

The NRO staff concludes that WEC has shown that the models used for the analysis of the shield building predict the observed experimental behavior and response with acceptable accuracy up to the design-basis SSE seismic load level. Also, the staff finds that the design has acceptable stress and strain values in the SC steel plates, tie bars, and studs. The staff also finds that WEC's adaptation of the ACI-349 Code for the design of the SC modules is acceptable. Finally, the staff finds the WEC's confirmatory analysis approach to be acceptable. We concur with the staff's conclusion.

The test specimens representing the SC modules with the closer tie-bar spacing used in regions of high out-of-plane demands failed in a ductile manner in all the tests. Some of the test specimens representing the SC wall modules with the tie-bar spacing used in the regions of low out-of-plane shear demands failed in a non-ductile manner in out-of-plane shear tests. This non-ductile behavior is the basis for a non-concurrence by an NRC staff member on the acceptability of the design of the shield building. In the view of the staff member, the behavior of the modules with increased tie-bar spacing is unacceptable. This non-concurrence was reviewed in both AP1000 Subcommittee and in full committee meetings.

As a matter of principle, structures important to nuclear safety should be designed so that, in the unlikely event the loads acting on the structure are larger than anticipated, the structure would behave in a ductile manner.

The staff member contends that this principle should be met by every element of the structure. WEC contends that it is the structure as a whole, not its elements, that ultimately matters, and that the design of the shield building does provide a structure that will behave in a ductile manner, because the low-ductility elements will approach their elastic limits only after those elements of the structure that do behave in a ductile manner have undergone significant plastic deformation. This approach is consistent with the intent of ACI-349, which requires ductile behavior only where demands are high and plastic deformation is expected to occur.

In the regions of low out-of-plane shear demands, the analysis shows that the out-of-plane shear capacity of the low ductility module is about 5 times greater than the applied shear load under design-basis loads. Indeed, except for some very small regions, the capacity is typically 10 times greater than the demand. Because the structural analysis follows typical seismic engineering practice and the finite element models used to describe the behavior of the SC models have been benchmarked to show satisfactory agreement with experiments even for loads greater than the design-basis loads, the NRO staff finds this margin to be acceptable, despite the uncertainties associated with any seismic analysis. We concur with the staff's conclusion. This conclusion is also consistent with the independent evaluation by the RES staff. All four of the consultants engaged by RES also agreed that the demand-to-capacity ratio was acceptable with sufficient margin. An additional expert consultant engaged by the ACRS, also agreed that margins were sufficient to ensure that the overall structural behavior was ductile.

The effort and scope of analysis and assessment required for the shield building in this case suggests that if SC composites are to be more widely used in nuclear applications, a consensus code should be developed, as has been done for other types of nuclear construction.

Analysis of Containment Vessel Cooling

The Passive Containment Cooling System is a safety-related system which is capable of transferring heat directly from the 130-foot diameter steel containment vessel (CV) to the environment. The Passive Containment Cooling System makes use of both the CV and the shield building surrounding the containment. A water distribution system, with two sets of weirs, is mounted on the outside surface of the steel CV and functions to distribute water flow on the containment exterior. The shield building directs natural draft air flow over the wetted exterior surface of the CV. The redesigned shield building reduces this air flow by about 20%, as compared to the existing, certified RC design.

Our review of WCAP-15846, Volume 1, Revision 1, “WGOTHIC Application to AP600 and AP1000,” revealed that the calculated time required to establish steady state coverage of the water film on the containment surface at prototypical flow rates was underestimated because of incorrect scaling of the 1/8 sector experimental result. This is non-conservative, inasmuch as a shorter time to reach steady state reduces the calculated peak containment pressure. WEC acknowledged the error and stated that correct scaling of the test data would result in a longer time to reach steady state film coverage at prototypical flow rates. However, WEC indicated that the analysis of record is based on an assumed value for the time to reach steady state coverage which is greater than that calculated using the correct scaling. Hence, the error should not impact the calculated peak containment pressure in the analysis of record. The staff should verify that the assumed time to reach steady state film coverage in the analysis of record is indeed longer than the corrected value obtained using the correct scaling.

Reactor Coolant Pump

The AP1000 utilizes four, hermetically sealed, high-rotational inertia, centrifugal canned-motor RCPs. The pump motor and all rotating components are contained inside a robust housing. The pumps circulate large volumes of high temperature, high pressure cooling water through the reactor vessel, loop piping, and steam generators.

In order to provide the rotational inertia necessary for flow coastdown, each pump uses two heavy flywheels of unique design. The flywheels contain high density tungsten alloy segments. A shrink fitting process uses a high strength retaining ring to hold the segments against a heavy-wall stainless steel inner hub. This retaining ring must resist all the centrifugal forces resulting from pump operation. The retaining ring is fabricated from a high strength 18% Cr, 18% Mn, iron based stainless steel (a material commonly used in electric generator applications but not in PWR primary coolant circuits). This assembly is seal-welded within a thin wall Alloy 625 (nickel base) cylindrical enclosure. The primary function of this enclosure is to isolate the tungsten segments and the retaining ring from the primary coolant surrounding the flywheel. After fabrication and inspection, the entire flywheel is then mated to the stainless steel pump shaft by a second shrink-fitting operation.

The design of the AP1000 pump makes it impractical (but not impossible) to perform periodic inservice inspection (ISI) of the Alloy 625 welds to assure that the enclosure remains leak tight. Providing assurance that the flywheel can operate without leaks for the 60-year life of the plant in the absence of ISI, is a daunting challenge. In the absence of a reliable leak detection method, our assessment is that the enclosure must be assumed to leak and that the retaining ring must be capable of operating in the primary water chemistry environment, and at temperatures at which the flywheel is designed to operate. The greatest threat to the integrity of the retaining ring is stress corrosion cracking (SCC).

If the retaining ring is susceptible to SCC, it can fracture after the cracks have reached a critical flaw size, releasing the heavy tungsten segments and causing rotor seizure. Such a seizure could have significant consequences, as discussed in Chapter 15 of the AP1000 DCD, Revision 17, including short term departure from nucleate boiling in the core, potential fuel failures, and offsite dose consequences. Because of the robustness of the pump housing, analysis has shown there is no significant risk of missiles from a flywheel failure exiting the pump.

WEC and the staff have stated that successful operation of the 18% Cr, 18% Mn retaining ring aterial in electric generator applications provides sufficient evidence to assure adequate SCC resistance in the flywheel application. We were not persuaded by this evidence. Electric generator environments are not prototypical of the PWR primary coolant environment. Further, no specific SCC nucleation or crack growth testing of the 18% Cr, 18% Mn retaining ring material has been performed to qualify the material for PWR service. We believe that the use of untested materials in such an important component as the RCP is fundamentally incompatible with General Design Criterion (GDC) 4. Consequently, we were concerned that adequate SCC resistance of the AP1000 flywheel retaining ring had not been demonstrated by testing in the primary water environment in which the flywheel is designed to operate.

WEC has responded to our concerns, and has stated that it will perform a test program to demonstrate the SCC resistance of the retaining ring material. The staff should incorporate this WEC commitment into the regulatory process, and should review the results of this testing with the Committee when available.

Flow Skirt

In the AP1000 DCD, Revisions 16 and 17, WEC proposed a change to its reactor internals. A flow skirt attached to the reactor vessel bottom head was added. The flow skirt is intended to provide a more uniform core inlet flow distribution and reduce the potential for excessive cross-flow, which could result in grid-to-rod fretting and fuel damage. We reviewed the effect of the flow skirt on core flow and flow distribution. Our review concluded that the addition of the flow skirt improves core inlet flow distribution and is satisfactory.

Human Factors Engineering

The staff review of HFE information included in the DCA was thorough, evaluating the HFE program, analyses, and design against the detailed guidelines of NUREG-0711. We are pleased that the four HFE DAC were closed as part of the DCA. This relieves substantial burden in the review of future combined license applications (COLAs). These four HFE DAC are listed below:

- Human Reliability Analysis is integrated with HFE design.
- Task Analysis is performed in accordance with the task analysis implementation plan.
- The human-system interface (HSI) design is performed for the Operation and Control Centers System in accordance with the HSI design implementation plan.
- An HFE program verification and validation implementation plan is developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan.

The staff review went well beyond the brief acceptance criteria stated in the DAC. For example, when the DAC required that a report exists that concludes the design is in conformance with the implementation plan, the review examined the content of the report, identifying omissions, incomplete analyses, and apparent errors through requests for additional information (RAIs) and open items. The review included staff audits of WEC analysis documents to ensure that all these issues were resolved.

Probabilistic Risk Assessment

The Probabilistic Risk Assessment (PRA) was completed as part of DCD Revision 15 and the most recent revision of the PRA Report is Revision 8 from 2007. DCD Revision 17, Chapter 19 includes very little new PRA information. During the staff review of DCD Revision 17, the staff performed an audit of the PRA at the WEC's headquarters. They reviewed changes to the PRA model that occurred after the submittal of the AP1000 PRA Report, Revision 8, including those related to RAIs and the amended design, as well as how the model had been converted from WEC's proprietary computer code to a more widely used linked-fault-tree code. The audit team explored the PRA by exercising the computer model and reviewing calculation notes documenting the bases for revisions to the PRA model that account for changes in the AP1000 design.

The audit team identified omissions and errors that were documented in open items that now have been closed. They found no other issues that required update of the DCD. In the audit report the staff reiterated their expectation that "before COLs begin to operate, they will develop plant-specific PRAs that conform to the appropriate revision and addenda of ASME/ANS-RA-S."

Digital Instrumentation and Control

The DCA submitted by WEC makes the following major changes to the DI&C System:

- Revised Chapter 7 to delete the use of the Eagle 21 System as an option for the Protection and Safety Monitoring System (PMS) and to provide for the use of the Common Q Platform as the microprocessor based computing platform in a DI&C architecture defined by WEC topical report WCAP-16675, "AP1000 Protection and Safety Monitoring System Architecture Technical Report."
- Revised the Diverse Actuating System (DAS) to be designed using field programmable gate arrays instead of a microprocessor based system.
- Revised the design of the Turbine Generator Overspeed Trip System from redundant, independent mechanical and electrical systems to redundant, independent electrical systems.
- Proposed the closure of DAC associated with the design requirements and system definition phases for the PMS and DAS, based on the more detailed descriptions of the designs provided in the DCA and referenced documents.

We completed a review of the proposed PMS architecture based on evaluating compliance with the four fundamental pillars of reliable DI&C microprocessor based system designs: redundancy, independence, deterministic processing behavior, and diversity and defense in depth (D3). Our review concluded that the redundancy and D3 pillars were met.

The staff found that the AP1000 design for DAS voting logic and diversity met the requirements and was acceptable. However, the staff identified that Automatic Depressurization System (ADS) spurious actuation could be a potential safety concern. WEC resolved this concern by making a change in the DCA to mitigate the potential for spurious ADS actuation. This resolution is acceptable.

During the review, it was noted that the watchdog timers were critical to ensuring that the independence criteria were met and that the PMS would actuate a reactor trip if all of the voting

processors in each division locked up due to a common cause failure (CCF). However, the design architecture for the watchdog timer operations was not clearly defined in the DCA or in referenced documents. Subsequently, WEC provided a detailed description of the watchdog timer design and operation. We consider this additional detail to be necessary and should be included in the DCA.
Our review of deterministic processing behavior noted that the topical report for the Common Q Platform identified that the bus loading in the processor should be limited to less than 70% of its capacity to ensure that deterministic processing was maintained. The DCD Tier 1 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the PMS did not include a test of the time response of the system from parameter input to control device actuation with the processor loaded to 70% of its capacity. WEC committed to including time response testing to verify system performance at maximum processor loading. We agree with this resolution, which should be reflected in the DCA.

Our review of the Turbine Generator Overspeed Trip System found that there was no specific test to confirm that the trip system would prevent exceeding 120% of rated speed as specified in the note following DCD Tier 2, Table 10.2.2, "Turbine Overspeed Protection." WEC identified two tests in DCD Chapter 14, (100% Load Rejection and Plant Trip from 100% Power) that will demonstrate that the Table 10.2.2 peak transient overspeed value of ≤108% is not exceeded. However, a review of those tests found that the performance criteria did not mention confirming the peak transient overspeed value, and WEC agreed to incorporate the ≤108 % in the performance criteria for these tests. This commitment should be included in the DCA.

The removal of the DAC resulting from these DI&C changes was evaluated by the staff. We agree with the staff resolutions for these DAC.

Diverse Actuating System Out of Service Limits

During the course of our review, we identified a concern which appears to apply to the existing certification, as well as to the proposed amendment. There are two actuation logic modes: automatic and manual. The automatic DAS logic mode functions to logically combine the automatic signals from the two redundant automatic systems on a two-out-of-two basis. The manual DAS is implemented by hard wiring the controls directly to the final loads, bypassing the normal path through the PMS and the DAS automatic logic. The manual DAS has a 30-day Technical Specification out of service (OOS) allowance and the automatic DAS has a 14-day investment protection reporting time for OOS time. The PMS Engineered Safeguards Features Actuation System (ESFAS) is a two-out-of-four system which is designed to fail as-is. The voting units for the system are the same microprocessor based units that are used for the reactor trip functions in the PMS. If a CCF locks up all of the voting units, the system fails as-is and will not perform a safeguards actuation if requested. The backup to PMS is the automatic and manual DAS. As presently specified, both of these backup systems are allowed to be OOS at the same time. If a safeguards action is requested while both are OOS, there is no backup available for independent actuation. We are concerned that allowing both automatic and manual DAS to be OOS at the same time results in an unnecessary and significant reduction in diversity of protection capability which is credited in the AP1000 PRA. Accordingly, we recommend that the staff seek commitments from COL holders to not allow both automatic and manual DAS to be OOS at the same time.

In summary, we agree with the staff's resolution of all of the open items for the AP1000 DCA with respect to the specific safety issues. The changes proposed in the AP1000 DCA maintain the robustness of the previously certified design. We conclude that there is reasonable assurance that the revised design can be built and operated without undue risk to the health and safety of the public. This conclusion is contingent on the results of our concurrent reviews of the aircraft impact assessment and long-term core cooling issues which will be discussed in separate letters.

Additional comments by ACRS Members Charles H. Brown Jr. and J. S. Armijo are presented below.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

Additional Comments by ACRS Members Charles H. Brown Jr. and J. S. Armijo

Squib Valve Post Seismic Testing

The Automatic Depressurization System (ADS) ADS-4 squib valves must operate to achieve post LOCA passive long-term cooling. They are actuated by an explosive charge and are one-time-use valves until the internals are replaced. Thus, once installed, they cannot be tested in service.

We asked if the entire valve was operationally tested after being subjected to qualification seismic testing. WEC stated NO, the basis being that the valves are extensively analyzed in accordance with ASME code requirements; motor operated valves (MOV) are not operationally tested after seismic testing; and the critical actuating parts, the charge and tension bolts, are individually tested after seismic testing in simulated prototype fixtures.

We do not agree with this position and recommend that they be operationally tested after seismic testing for the following reasons:

1. Failure of the ADS-4 squib valves due to an unknown common cause mechanism prevents initiation of post LOCA passive long-term cooling.

2. This is a first time application for this service in nuclear power plants.

3. The valve actuation is a one-time pulse that ignites a charge, pushes a piston through the range of a cylindrical channel to rupture a shear cap causing the released cap to rotate about a pin to allow flow to occur. The only force to push the shear cap out of the way other than gravity is the pressure of the fluid. If seismic forces warp the channel, inhibiting or reducing piston travel; or warp the shear cap such that the shear cap does not break cleanly; or bend the pin preventing rotation of the valve disk, then the valve becomes non-operational.

4. An MOV is not a valid basis for comparison since it has a torque applying continuous force to drive a valve open or shut.

5. While an analysis for this unique valve is useful to assess the potential of the design to pass the post seismic test, it has not been validated as being satisfactory for full qualification without actual post seismic qualification operational testing.

Additional Amplifying Discussion

The ability to achieve satisfactory post LOCA passive long-term cooling has been extensively analyzed and tested in excruciating detail relative to types of debris, particulates, chemistry, and environment temperature to ensure sump and other screens do not become clogged. In our opinion, it is incongruous to now conclude that the valves critical to ensuring post LOCA passive long-term cooling will perform satisfactorily without post seismic qualification prototypical operational testing.

REFERENCES

1. U.S. Nuclear Regulatory Commission, “Advanced Copy of the Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” (ML103260072)

2. Letter to U.S. Nuclear Regulatory Commission, “Westinghouse Application to Amend the AP1000 Design Certification,” APP-GW-GL-700, Revision 16, 05/26/2007 (ML071580757)

3. Letter to U.S. Nuclear Regulatory Commission, “Update to Westinghouse’s Application to Amend the AP1000 Design Certification Rule,” APP-GW-GL-700, Revision 17, 09/22/2008 (ML083220482)

4. Westinghouse, AP1000 Design Control Document (DCD), APP-GW-GL-700, Revision 18, December 1, 2010, APP-GW-GL-700, Revision 18

5. NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design (NUREG-1793),” 09/2004 (ML043450344, ML043450354, ML043450284, ML043450290, and ML043450274)

6. NUREG-1793 Supplement 1, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design” 12/2005 (ML060330557)

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APPENDIX

CHRONOLOGY OF THE ACRS REVIEW OF THE WESTINGHOUSE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

The extensive ACRS review of the AP1000 DCD and its interactions with representatives of the NRC staff and Westinghouse are discussed in the minutes and transcripts of the following ACRS meetings.

ACRS MEETING/DATES	SUBJECT
562nd ACRS Meeting 5/7-9/2009	Status and Update Concerning Revisions to the AP1000 Design Control Document
AP1000 Subcommittee 7/23-24/2009	AP1000 DCD and NRC Staff's AFSER for Chapters 1, 4, 5, 10,11,12,14, 16, 17, and 19
AP1000 Subcommittee 10/6-7/2009	AP1000 DCD and NRC Staff's AFSER for Chapters 3, 8, and 18
AP1000 Subcommittee 11/19-20/2009	AP1000 DCD and NRC Staff's AFSER for Chapters 7 and 9 Long-Term Core Cooling
AP1000 Subcommittee 2/2-3/2010	AP1000 DCD and NRC Staff's AFSER for Chapter 15 Gas Intrusion Loss of Large Areas Regulatory Treatment of Non-Safety Systems RCP Issues
AP1000 Subcommittee 4/22/2010	Loss of Large Areas RCP Materials Elbow Taps Screening Criteria for Thermal Striping High-Density Polyethylene Connections Shield Building

ACRS MEETINGS/DATES	SUBJECT
AP1000 Subcommittee 6/24-25/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 4, 10, 11, 12, 14, and 22
AP1000 Subcommittee 7/21-22/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 2, 3, 16, and 17
AP1000 Subcommittee 9/20-21/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 5, 7, 8,13, and 18 AP1000 Containment Corrosion Prevention
AP1000 Subcommittee 10/5/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 6 and 15 Long-Term Core Cooling
AP1000 Subcommittee 11/2-3/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 9 and 19 Aircraft Impact Assessment
577th ACRS Meeting 11/4-6/2010	Long-Term Core Cooling
AP1000 Subcommittee 11/17-19/2010	AP1000 DCD and NRC Staff's AFSER for Chapters 3,15, and 23 Shield Building Issues Long-Term Core Cooling Aircraft Impact Assessment
AP1000 Subcommittee 12/1/2010	Action Items
578th ACRS Meeting 12/2-4/2010	Final ACRS Review of the AP1000 DCD

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINTON, DC 20555 - 0001

December 20, 2010

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: LONG-TERM CORE COOLING FOR THE WESTINGHOUSE AP1000 PRESSURIZED WATER REACTOR

Dear Chairman Jaczko:

During the 577th and 578th meetings of the Advisory Committee on Reactor Safeguards (ACRS), November 4-6, and December 2-4, 2010, we reviewed the NRC staff's safety evaluation of the adequacy of long-term core cooling as it applies to the AP1000 design certification amendment application. AP1000 long-term core cooling performance was also reviewed during subcommittee meetings held on November 19-20, 2009, October 5, November 17-19, and December 1, 2010. During these meetings, we had the benefit of discussions with representatives of the NRC staff and the Westinghouse (WEC or applicant). We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. The regulatory requirements for long-term core cooling for design basis accidents have been adequately met, and the issue is closed for the AP1000 design.

2. This conclusion is based on the cleanliness requirements specified in the amendment. Any future proposed relaxation of these requirements will require substantial additional data and analysis.

BACKGROUND

On May 8, 2008, the Commission issued a Staff Requirements Memorandum (SRM) stating that, "the ACRS should advise the staff and Commission on the adequacy of the design basis long-term core cooling approach for each new reactor design based, as appropriate, on either its review of the design certification or the first license application referencing that reactor design." The main focus of the Commission's concern was the ability of the safety systems to provide adequate core cooling over extended time periods when the Emergency Core Cooling System (ECCS) recirculation mode is activated during a design basis accident (DBA).

The AP1000 is a pressurized light water reactor design that incorporates new passive safety features not found in current operating pressurized water reactors (PWRs). These include a Passive Containment Cooling System (PCS) to transport heat to the ultimate heat sink for accident scenarios.

Many aspects of long-term cooling (LTC), excluding the effects of debris, were considered as part of the AP1000 certification process that was completed in January 2006. This letter report addresses the effect of debris on LTC.

DISCUSSION

For AP1000 LTC, coolant is driven by gravity head through the core. The coolant exits, as a steam-water mixture, mainly through the Automatic Depressurization System (ADS-4) valves. The steam flowing out from the core removes decay heat and is condensed on the inside of the steel containment shell. The condensed water flows down the containment walls, is collected in the In-Containment Refueling Water Storage Tank (IRWST), and is recirculated. Screens placed between the IRWST and the core capture debris. Sump screens are placed in another possible flow path, which is through the loop compartment to the core.

During loss-of-coolant accidents (LOCAs) the level in the IRWST tank drops, redistributing water to the region around the reactor vessel and associated piping, causing much of the piping to be submerged. Breaks in this piping, such as in the cold legs or the direct vessel injection (DVI) lines, can be submerged and provide an unfiltered flow path to the reactor core.

The main sources of debris are: 1) latent containment debris, such as hair and clothing fibers; 2) debris generated by LOCA jets and exposure to post LOCA conditions; and 3) chemical precipitates that form in the recirculating water stream. WEC has taken advantage of what has been learned with regard to the GSI-191 issue for the fleet of operating PWRs. Efforts have been made in the design to minimize LOCA-generated debris by selecting low fiber, low particulate insulation and LOCA resistant qualified coatings. Stringent containment cleanliness requirements have been imposed in the amendment that limit fibrous latent debris and the amount of aluminum that can be submerged. Sump screens have been designed to assure negligible reduction in recirculation flows due to debris accumulation on them.

Because of these actions, any potential problems with LTC would primarily be due to flow blockage in the core which may trap materials that pass through the screens, and more importantly, materials that enter the core directly through submerged breaks. The possibility that unfiltered water, carrying in some cases a major portion of the suspended fibrous and particulate debris, will gain ingress directly to the core is unique to the AP1000 design. Furthermore, the gravity head available in the AP1000 for driving flow through a core in which debris has accumulated is limited. Both of these factors add to the difficulties in determining the adequacy of AP1000 LTC.

In the certified design, the applicant carried out a series of calculations using WCOBRA/TRAC, which had been accepted for analysis of LTC, without considering debris. Resolution of debris effects was deferred to the combined license (COL) stage but is now being addressed in the amendment. In the calculations for the design certification amendment, WCOBRA/TRAC was also used. The effect of debris, which mainly causes in-vessel head losses, was modeled by introducing a constant loss factor at the core inlet. The purpose of these calculations was to determine how the loss factor affected ADS-4 vent qualities (the mass fraction of steam), pressure loss across the debris bed, and mass flux through the core. Based on analysis of the results, the applicant proposed what is effectively an acceptance criterion that requires pressure drop through the debris bed to be less than a specified amount at a specified flow rate. When the criterion was met, the WCOBRA/TRAC results indicated that the ADS-4 vent quality would be less than 50 percent which resulted in acceptable boron concentration. At our request, additional results were obtained with higher loss factors to elucidate the margins inherent in the proposed acceptance criterion with regard to critical heat flux and boron concentrations. These indicated sufficient margin to account for uncertainties, and we agree that the acceptance criterion should be as proposed by the applicant.

To determine whether blockage under representative debris loadings and flow conditions would meet the acceptance criterion, the applicant conducted a series of tests in a pumped flow loop. The loop incorporated a part-length fuel bundle with representative inlet and spacer geometries. Flow rates were varied to simulate the transient mass flux through the core as the debris bed built up, though the lowest flow rates studied were somewhat higher than the value of the flow rate for the acceptance criterion. Fibrous and particulate debris loadings were conservative but were varied over a narrow range. An approved surrogate material was added over a period of time to simulate the effect of chemical precipitates, such as aluminum oxyhydroxide that might form. The reference experimental protocol was selected to follow the sequence of events expected for the long term recirculation phase of a DBA. However, the exact protocol that should be used is unclear, and tests have shown that variations in protocol can result in significant differences in pressure losses. For example, in a test where the protocol was inadvertently varied to follow a non-representative event sequence, a significantly larger debris-bed pressure loss was obtained than for the same case run with the reference protocol. However, the pressure loss still remained within acceptable limits.

For the tests used to determine whether the acceptance criterion could be met, the fibrous debris used was derived from NUKON insulation which may not be typical of the latent debris that might accumulate in the core in an AP1000 DBA. Two tests were conducted with non-reference protocols using debris containing hair and clothing fibers. While the pressure loss behavior was somewhat different from that observed in the NUKON-based test, the pressure losses were within the acceptable range.

Most of the tests were conducted at room temperature. In two exploratory tests, debris-bed pressure losses decreased significantly when the temperature was raised to values closer to those expected during LTC. The lower pressure losses are consistent with the effect of increasing temperatures on water viscosity. However, the net effect of increased temperature on head loss is still uncertain since organic materials may behave differently at LTC temperatures than the NUKON-based debris used in the tests. Absent additional experiments at LTC temperatures using organic fibers (hair, clothing) and prototypical water chemistry, it is not certain that the observed benefit of higher temperature will provide additional margin.

In the tests, the head losses that arose from debris accumulation in the fuel inlet region were rather low when the debris consisted of fibers and particulates alone. However, when the surrogate chemical precipitates were added gradually, head losses rose sharply initially, but generally leveled off as more was added. The effect of the chemical precipitates will depend on the rate of their formation. Although this is uncertain, the rate at which surrogates were added in the tests appears to be conservative.

Radiolysis in the containment atmosphere and doses to cable insulation might form small amounts of nitrogen oxides and hydrogen chloride which may acidify the water condensed on the containment wall. The acidified water may leach zinc from the containment coating. If some zinc does dissolve into the recirculating water stream, the chemical load that should be considered in evaluating debris head losses would be increased. While the experiments indicating that head losses level off with the addition of chemical surrogates suggest that the effect of the possible zinc load could be small, the effect has not been investigated and adds to the uncertainties.

In view of the relatively narrow range of conditions explored in the applicant's test program and the significant uncertainties in the results, a site visit was conducted to better understand the AP1000 related results in the context of in-core debris effects found in the PWR Owner's Group (PWROG) experiments. These cover a wide range of conditions and, while not directly applicable to the AP1000, offer valuable insights into the effects of various experimental parameters.

As a result of the issues arising in subcommittee meetings and the site visit, additional experimental results from the PWROG program at lower flow rates and higher fiber loadings were made available to us.

When the additional WCOBRA/TRAC analyses and the additional experimental results are taken into account, in-core debris bed pressure losses appear to meet the acceptance criterion with sufficient margin to account for the uncertainties, including those due to chemical effects, experimental protocol, and debris constituents. This conclusion is based on the limits on latent debris and submerged aluminum specified in the amendment. These cleanliness specifications should not be relaxed without additional analyses, a much wider range of experiments at prototypical conditions, and NRC review of these findings.

In summary, debris generation during DBAs has been minimized by the choice of LOCA-resistant insulation and coatings. This, together with the large flow area sump screens, results in negligible head losses except in the inlet regions of the core. With regard to in-vessel debris effects, the acceptance criterion established by the applicant is adequate to assure LTC. The criterion is met with sufficient margin to account for uncertainties provided the stringent cleanliness requirements specified in the amendment are maintained. The AP1000 design, therefore, meets the regulatory requirements for LTC during design basis accidents.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

References:

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2. Letter to U.S. Nuclear Regulatory Commission, Submittal of APP-GW-GLE-002 Revision 7 – “Impacts to the AP1000™ to Address Generic Safety Issue (GSI)–191,” 07/13/2010 (ML101970030)

3. Westinghouse Technical Report, WCAP-17028-P Revision 6, “Evaluation of Debris-Loading Head-Loss Tests for AP1000™ Fuel Assemblies During Loss of Coolant Accidents,” 06/29/2010 (ML102030227; ML102030219; ML102030221; ML102030222; ML102030223; ML102030215)

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5. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents,” 02/26/2010 (ML100640574)

6. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents, Enclosure 21,” 02/26/2010 (ML100640578)

7. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents, Enclosure 22, APP-GW-GLR-092 Proprietary Rev. 0,” 02/26/2010 (ML100640585)

8. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents, Enclosure 25, APP-GW-GLR-110 Rev. 0 Proprietary,” 02/26/2010 (ML100640586)

9. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of Technical Report APP-GW-GLR-079 Revision 8 (TR-026), (Proprietary & Non-Proprietary) “AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA” Enclosure 3, 07/20/2010 (ML102170124)

10. Letter to U.S. Nuclear Regulatory Commission, “Submittal of APP-GW-GLE-002 Revision 7 - Impacts to the AP I000™ to Address Generic Safety Issue (GSI)-191,” 12/13/2010 (ML101970030)

11. Letter to U.S. Nuclear Regulatory Commission, “AP1000 Response to Request for Additional Information (SRP6.2.2),” Enclosure, 07/30/2010 (ML102160216)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

January 19, 2011

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE AIRCRAFT IMPACT ASSESSMENT FOR THE WESTINGHOUSE AP1000 DESIGN CERTIFICATION AMENDMENT APPLICATION

Dear Chairman Jaczko:

During the 579th meeting of the Advisory Committee on Reactor Safeguards, January 13-15, 2011, we reviewed the staff's Safety Evaluation Report (SER) on the Aircraft Impact Assessment (AIA), which is part of the Westinghouse (WEC or the Applicant) AP1000 Design Certification Amendment (DCA) application. Our AP1000 subcommittee held meetings on November 2-3, November 17-19, and December 15-16, 2010, and reviewed the staff's SER and AIA inspection report. During these meetings, we had the benefit of discussions with representatives of the NRC staff and WEC. The AIA was made available to us by the applicant for review prior to our AP1000 subcommittee meeting of November 2-3, 2010. We also had the benefit of the documents referenced. This letter fulfills the requirement of 10 CFR 52.53 that the ACRS report on those portions of the application which concern safety.

CONCLUSION AND RECOMMENDATION

The WEC AIA for the design described in the AP1000 DCA application, as modified to resolve NRC inspection findings, complies with the requirements of 10 CFR 50.150. Analyses show that the containment remains intact following the impact of a large commercial aircraft. The reactor core remains cooled, and spent fuel pool integrity is maintained.

The staff should evaluate information and analyses presented to the ACRS, but not subjected to staff review or inspection, to determine if there is a need for further revision of the design control document (DCD), or a need for further inspections.

BACKGROUND

The results of the AP1000 AIA are a part of the AP1000 DCA application. The AP1000 design was previously certified and the existing AP1000 certification rule references DCD Revision 15. DCD Revision 18 was submitted by WEC in a letter dated December 1, 2010, and it incorporates changes in Revision 16, submitted on May 26, 2007; in Revision 17, submitted on September 22, 2008; as well as those changes made subsequent to the submittal of Revision 17, which are identified in Chapter 23 of the Advanced Final Safety Evaluation Report. We held a series of meetings with the NRC staff and the applicant on the AP1000 DCA application. We wrote a letter, dated December 13, 2010, following our review of the amendment. Our assessment of the AP1000 AIA was not included in the letter.

As required by 10 CFR 50.150, applicants for new nuclear power plants must perform an assessment of the effects of the impact of a large, commercial aircraft. Using realistic analyses, applicants must identify and incorporate into the facility those design features and functional capabilities needed to show that, with reduced use of operator action; (1) the reactor core remains cooled or the containment remains intact, and (2) spent fuel cooling or spent fuel pool integrity is maintained (referred to as the acceptance criteria). Applicants are required to submit a description of the design features and functional capabilities relied upon in the AIA and a description of how these features and capabilities ensure that the acceptance criteria are met. Since the impact of a large, commercial aircraft is a beyond-design-basis event, applicants may use non-safety-related features or capabilities to satisfy the requirements of 10 CFR 50.150.

From September 27, 2010, through October 1, 2010, the staff conducted an inspection of the WEC AP1000 AIA. Based on the results of this inspection, the staff determined that NRC requirements had not been fully met. The inspection revealed that WEC did not use realistic analyses for certain aspects of its AIA and did not fully identify and incorporate into the DCD those design features and functional capabilities credited. WEC responded to the inspection report and proposed corrective actions in its letter to the NRC dated November 12, 2010. The staff issued a letter, dated November 23, 2010, stating that the proposed corrective actions were satisfactory. The staff may review the implementation of the corrective actions during a future inspection to determine that full compliance has been achieved and maintained.

DISCUSSION

The AIA performed by the applicant uses the industry guidance in NEI 07-13, Revision 7, endorsed in Draft Regulatory Guide DG-1176. The results of the AIA show that the modified AP1000 design, described in the application, meets the acceptance criteria of the AIA rule by maintaining containment integrity and spent fuel pool integrity.

The key AP1000 design features identified by WEC to satisfy the requirements of 10 CFR 50.150 include: presenting a small target with a reduced set of safety-related structures, systems, and components (SSCs); a redesigned shield building which protects the steel containment vessel from penetration due to impact[8]; simplified, passive safety equipment for core cooling; no active equipment required for spent fuel pool cooling; and redundancy and defense-in-depth in equipment design. In accordance with 10 CFR 50.150, WEC provided an assessment in the respective technical areas of structures, reactor systems, fire, and shock.

For the structural assessment, WEC used the impulse curve supplied by the NRC and the finite element analysis code LS-DYNA. All of the aircraft strikes analyzed using this code was on the shield building. The redesigned shield building, using a modular, steel concrete composite (SC) structure, reduces passive heat removal air flow. The effects of air flow reduction on containment integrity during accidents were analyzed and shown to be acceptable. Based on the results of the assessment, WEC concluded, and the staff agreed, that both the containment and spent fuel pool remain intact and that core and spent fuel cooling are maintained.

During our November 2-3, 2010 AP1000 subcommittee meeting, we questioned whether the worst-case locations for aircraft impact had been considered. WEC addressed this issue during our November 17-19, 2010, AP1000 subcommittee meeting.

The AP1000 shield building includes a 32 ft. diameter opening in the conical roof which is an essential feature of the passive containment cooling design. This opening is surrounded by the Passive Containment Cooling System water storage tank. During our November 2-3, 2010, subcommittee meeting, issues arose concerning the potential for significant aircraft impact debris to pass through the opening and impact the steel containment vessel. WEC conducted appropriate analyses, which we reviewed during our November 17-19, 2010, subcommittee meeting. Using realistic assumptions for the impact locations of concern, these analyses demonstrated that no significant debris would impact the steel Containment Vessel (CV). In addition, WEC performed a more conservative analysis in which a large mass consisting of debris and the shield plate, was assumed to fall on the steel CV. This impact resulted in only a relatively small amount of plastic deformation and no penetration of the CV.

Our December 13, 2010, letter concerning the AP1000 DCA application describes the SC design, including the addition of tie bars between opposite faceplates of the SC modules. The spacing of these tie bars is smaller in areas of higher, out-of-plane, design basis shear demands - i.e., near discontinuities and connections - than it is in the majority of the shield building wall structure where these demands are lower. Aircraft impacts, unlike design basis events, can impart high out-of-plane shear demands in regions of the shield building wall with greater tie bar spacing. As discussed in our letter of December 13, 2010, these areas can fail in

[8] The shield building redesign is discussed in our letter dated December 13, 2010.

a non-ductile manner under such loads. In order to assure acceptable realism in the analyses, it must be demonstrated that the finite element models used in the AIA adequately describe this non-ductile behavior under high out-of-plane shear loads. WEC provided comparisons of the predictions of the LS-DYNA model with an experiment on a beam representing a SC structure with greater tie bar spacing under high out-of-plane shear loads. The load-deformation behavior predicted by the model agreed well with the results of the experiment; the comparison adequately supports the use of the model for these analyses.

In addition to the possibility of global structural failure, there is also a potential for local failure due to penetration by hard objects such as an engine or landing gear. The AIA analysis included comparisons of the predictions of the LS-DYNA model with penetration tests conducted in Japan on SC structures. The predictions show adequate agreement with the tests. Although the geometry of the specimens in these tests differs from that of the shield building, the comparisons support the use of the model to predict local failures associated with aircraft impact.

WEC demonstrated that AIA requirements with respect to core and spent fuel cooling are met. This is because the systems required for design basis core cooling are located inside containment, which is protected by the redesigned shield building, and there are no active systems required for cooling of spent fuel. In addition, WEC demonstrated that at least one backup water source is always available for cooling.

Similarly, for the fire aspect of AIA, based on the limited systems required for core cooling in the AP1000, and their location within the intact containment, WEC demonstrated that the requirements of 10 CFR 50.150 are met.

Finally, with regard to the effects of shock associated with aircraft impact, WEC demonstrated that these shock loadings are less than those resulting from a design basis seismic event.

The AP1000 AIA was reviewed in parallel with the development of DCD Revision 18, which was submitted on December 1, 2010. Also, the staff conducted an inspection of the AIA and resolved their findings with WEC, as described in a letter dated November 23, 2010. In parallel with these activities, we conducted subcommittee meetings to review the AIA during which WEC responded with information and analyses, some of which may not be reflected in the DCD, as revised, or within the scope of the staff's inspection. In view of these parallel activities, the staff should evaluate information and analyses presented to the ACRS, but not subjected to staff review or inspection, to determine if there is a need for further revision of the DCD, or a need for further inspections.

The AIA for the design described in the AP1000 DCA application, as modified to resolve the staff's inspection findings, complies with the requirements of 10 CFR 50.150. Following the impact of a large commercial aircraft, the containment remains intact, the reactor core remains cooled, and spent fuel pool integrity is maintained.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Advanced Copy of the Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design" various dates 2010 (ML103260072)
2. Letter to U.S. Nuclear Regulatory Commission, "Westinghouse Application to Amend the AP1000 Design Certification," APP-GW-GL-700, Revision 16, May 26, 2007 (ML071580757)
3. Letter to U.S. Nuclear Regulatory Commission, "Update to Westinghouse's Application to Amend the AP1000 Design Certification Rule," APP-GW-GL-700, Revision 17, September 22, 2008 (ML083220482)
4. Westinghouse, AP1000 Design Control Document (DCD), APP-GW-GL-700, Revision 18, December 1, 2010 (ML103480059 and ML103480572)
5. NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design (NUREG-1793)" September 2004 (ML043450344, ML043450354, ML043450284, ML043450290, and ML043450274)
6. NUREG-1793, Supplement 1, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," December 2005 (ML060330557)
7. ACRS letter to the NRC Chairman on the AP1000 DCD amendment review, December 13, 2010 (ML103410351)
8. NRC Letter to WEC on "Ap1000 Pressurized Water Reactor Design Aircraft Impact Assessment Inspection, NRC Inspection Report No. 05200006/2010-203 and Notice of Violation," October 28, 2010 (ML10298058311)
9. WEC response to NRC on "Reply to Notice of Violation Cited in NRC Inspection Report No.: 05200006/2010-203 dated October 28, 2010," November 12, 2010 (ML1032104091)
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11. NRC Letter, "Aircraft Impact Assessment for New Reactor Designs," May 17, 2007 (ML071360212)
12. NRC Letter, "Issuance of Order Imposing Safeguards Information Protection Requirements and Fingerprinting and Criminal History Records Check Requirements for Access to Safeguards Information," September 12, 2007 (ML072220401)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 19, 2011

Mr. R.W. Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO THE FEBRUARY 5, 2011, EDO LETTER REGARDING THE FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE AMENDMENT TO THE AP1000 DESIGN CONTROL DOCUMENT

Dear Mr. Borchardt:

During the 583rd meeting of the Advisory Committee on Reactor Safeguards, May 12 - 14, 2011, we reviewed your February 5, 2011, letter responding to our December 13, 2010, letter regarding the staff's final Safety Evaluation Report associated with the amendment to the AP1000 Design Control Document (DCD). In our December 13, 2010, letter, we expressed concern that the potential for failure of the reactor coolant pump (RCP) flywheel due to stress corrosion cracking (SCC) should be addressed by demonstrating that the material used is qualified for the primary water environment in which the flywheel is designed to operate. Your letter states that the staff believes this qualification testing is unnecessary because the safety consequences of a RCP flywheel failure have been adequately addressed by designing the pump casing to contain any potential missiles.

Westinghouse has committed to qualification testing as discussed further below. However, we also wish to respond to the rationale provided for the staff's determination that qualification testing of the flywheel material is unnecessary. As noted in our letter, a rotor seizure resulting from flywheel failure "could have significant consequences, as discussed in Chapter 15 of the AP1000 DCD, Revision 17, including short term departure from nucleate boiling in the core, potential fuel failures, and offsite dose consequences." The potential for these effects of a locked rotor accident, and the dynamic forces which would result at the bolted connection of the RCP to the primary system, should be minimized by using flywheel material which has been qualified to be resistant to SCC in the primary system.

Requiring use of available means to reduce the potential for a locked rotor event by using qualified flywheel material is warranted notwithstanding the fact that the flywheel is intended to be protected from exposure to the primary coolant by a surrounding Alloy 625 enclosure, because the integrity of the enclosure is not subject to periodic in-service inspection and therefore cannot be assured.

We have received a copy of the stress corrosion test program to be performed by Westinghouse to demonstrate the SCC resistance of the AP1000 RCP flywheel retaining ring material. We are concerned with the ability of the test program to provide reasonable assurance that the material will be resistant to SCC in the primary coolant environment. Our specific concern is with the proposed use of elastically loaded bent beam samples to demonstrate resistance to the initiation of stress corrosion cracks. This test method has been found to be unreliable for all but highly susceptible materials. For example, testing of hundreds of bent beam specimens for thousands of hours by the General Electric Company in the early 1960s failed to predict the susceptibility of welded Type 304 stainless steel components to SCC in boiling water reactors (BWRs). The crack growth rate (CGR) tests proposed by Westinghouse can provide a sensitive assessment of susceptibility, but the test protocols are not easily standardized. Slow strain rate tests (SSRT) demonstrated SCC susceptibility for BWR environments consistent with in-reactor performance. Today, the SSRT method is widely used to demonstrate resistance to SCC initiation, and a standard protocol is available (ASTM G129-00). Passing this test provides a high degree of assurance that a material is highly resistant to SCC initiation, and SSRT are generally easier and quicker to perform than CGR tests. Furthermore, we consider SSRT to be the most appropriate method for demonstrating SCC resistance of the retaining ring material.

In our December 13, 2010, letter, we also identified a concern that allowing both the automatic and manual modes of actuation of the Diverse Actuation System (DAS) to be out of service at the same time would result in an unnecessary and significant reduction in diversity of the protection capability, which is credited in the AP1000 probabilistic risk assessment (PRA). Thus, we recommended that the staff seek commitments from combined license holders to not allow both automatic and manual DAS to be out of service at the same time. Following review of the staff's response to our letter, we continue to make this recommendation for all the reasons enumerated in our letter. Some compensatory actions should be taken, if both automatic and manual DAS are out of service.

While we understand the logic described by the staff, common cause failure of the Protection and Safety Monitoring System is poorly understood, and no credible reliability models or data are available. Therefore, there is substantial unquantified uncertainty in the PRA results used to evaluate the importance of DAS. We consider both automatic and manual DAS as defense-in-depth measures against a poorly understood set of "common cause" failure mechanisms that could disable a reactor trip. To ensure that the defense-in-depth role is fulfilled, unavailability of manual DAS should be minimized, limited to on the order of no more than 72 hours. The current limiting condition for operation on manual DAS of 30 days is too long. This is in addition to requiring compensatory action in the event that both automatic and manual DAS are out of service, as indicated above.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

References:

1. Letter to Chairman Jaczko, AP1000 DCD Amendment Review, 12/13/2010, (ML103410351)

2. Package: NRC EDO letter, Report on the Final Safety Evaluation Report Associated with the Amendment to the AP1000 Design Control Document, 02/05/2011, (ML103560411)