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Final Safety Evaluation Report

Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design

Volume 2 (Chapters 4 – 8)

Office of New Reactors

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ABSTRACT

This final safety evaluation report documents the technical review of General Electric-Hitachi's (GEH's) Economic Simplified Boiling-Water Reactor (ESBWR) design certification. GEH submitted its application for the ESBWR design on August 24, 2005, in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52. The NRC formally docketed the application for design certification (Docket No. 52-010) on December 1, 2005.

The ESBWR design is a boiling-water reactor (BWR) rated up to 4,500 megawatts thermal (MWt) and has a rated gross electrical power output of 1,594 megawatts electric (MWe). The ESBWR is a direct-cycle, natural circulation BWR that relies on passive systems to perform safety functions credited in the design basis for 72 hours following an initiating event. After 72 hours, non-safety systems, either passive or active, replenish the passive systems in order to keep them operating or perform post-accident recovery functions directly. The ESBWR design also uses non-safety-related active systems to provide defense-in-depth capabilities for key safety functions provided by passive systems. The ESBWR standard design includes a reactor building that surrounds the containment, as well as buildings dedicated exclusively or primarily to housing related systems and equipment.

On the basis of its evaluation and independent analyses, as set forth in this report, the NRC staff concludes that GEH's application for design certification meets the requirements of 10 CFR Part 52, Subpart B, that are applicable and technically relevant to the ESBWR design. Appendix F includes a copy of the report by the Advisory Committee on Reactor Safeguards, as required by 10 CFR 52.53.

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4.0 REACTOR

4.1 Introduction

In the economic simplified boiling-water reactor (ESBWR) design control document (DCD) Tier 2, Revision 9, Chapter 4 the mechanical components of the ESBWR reactor and reactor core, including the fuel system design (fuel rods and fuel assemblies), nuclear design, thermal-hydraulic design, reactor materials, and functional design of the control rod drive (CRD) system are described.

DCD Tier 2, Revision 9, Chapter 4, also identifies certain areas as “Tier 2*” information, departures from which require prior approval from the staff of the U.S. Nuclear Regulatory Commission (NRC). Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” Appendix A, “Design Certification Rule for the U.S. Advanced Boiling Water Reactor,” Section VIII.B(6.a), provides a definition and the criteria governing Tier 2* information.

The following sections in DCD Tier 2, Revision 9, Chapter 4, include Tier 2* information:

- Section 4.2.7
- Section 4.3.6
- Section 4.4.8
- Appendix 4A
- Appendix 4B
- Appendix 4C

4.2 Fuel System Design

The fuel system comprises the fuel assembly and the reactivity control assembly. The fuel assembly consists of the full-length and part-length fuel rods, grid spacers, water rods, upper and lower tie plates, and the channel. DCD Tier 2, Revision 9, Appendix 4B defines the fuel licensing acceptance criteria that must be satisfied by any fuel design to be loaded into the ESBWR core. DCD Tier 2, Revision 9, Appendix 4C defines the control rod acceptance criteria that must be satisfied for any control rod design used in the ESBWR core.

4.2.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 4.2, Appendix 4B, and Appendix 4C in accordance with the regulatory guidance for the review of fuel system design, including adherence to applicable general design criteria (GDC) discussed in NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (hereafter referred to as the SRP), Section 4.2, Draft Revision 3, issued June 1996. The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, generic issues (GI), bulletins (BL), generic letters (GL), or technically significant acceptance criteria (except Appendix 4B, Interim Criteria and Guidance for the reactivity initiated accidents) beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 3 of SRP Section 4.2, issued in June 1996, is acceptable for this review.

The following GDCs from 10 CFR Part 50, Appendix A and regulations are applicable in SRP Section 4.2:

- GDC 10, “Reactor design,” as it provides assurance that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs);

- GDC 27, “Combined reactivity control systems capability,” as it relates to the combined effect of the reactivity control system being designed with appropriate margin and capability to control reactivity changes while at the same time maintaining the capability to cool the core;

- GDC 35, “Emergency core cooling,” as it relates to emergency core cooling so that following any loss of reactor coolant, 1) fuel and clad damage that could interfere with core cooling is prevented, and 2) clad metal-water reaction is limited to negligible amounts.

- 10 CFR 50.46 as it relates to the cooling performance analysis

- 10 CFR 52.47(b)(1) which requires that a DC application contain the proposed ITAAC

In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance of the following:

- The fuel system is not damaged as a result of normal operation and AOOs.

- Fuel system damage is never so severe as to prevent control rod insertion when it is required.

- The number of fuel rod failures is not underestimated for postulated accidents.

- Coolability is always maintained.

The staff reviewed the Tier 1, Tier 2, and Tier 2* fuel design and control rod design acceptance criteria to ensure that the requirements outlined in SRP Section 4.2 are satisfied.

The DCD requirements for the contents of applications appear in 10 CFR 52.47. SRP Section 14.3.4 provides guidance related to the approval status of fuel system design and the designation of DCD requirements, including the following:

- The specific fuel, control rod, and core designs presented in Tier 2 will constitute an approved design that may be used for the combined operating license (COL) first-cycle core loading without further staff review. If any other core design is requested for the first cycle, the COL applicant or licensee must submit for staff review the specific fuel, control rod, and core design analyses as described in DCD Tier 2, Revision 9, Chapters 4, 6, and 15. Much of the detailed supporting information in Tier 2 for the nuclear fuel, fuel channel, and control rod, if considered for change by a COL applicant or licensee referencing the certified standard design, would require prior NRC approval. Therefore, for the evolutionary designs, the staff concluded that this information should be designated as Tier 2* information. However, the staff allowed some of the Tier 2* designation to expire after the first full-power operation of the facility, when the detailed design would be complete and the core performance characteristics would be known from the startup and power ascension test programs. The NRC bears the final responsibility for designating which material in Tier 2 is Tier 2*.

- Inspections, tests, analyses, and acceptance criteria (ITAAC) are not required for Tier 1 information in the fuel, control rod, and core design areas because of the requirement for prior NRC approval of any proposed changes to the approved design.

- Post-fuel-load testing programs (e.g., startup testing and power ascension testing) verify that the actual core performs in accordance with the analyzed core design.

Only fuel assembly and control rod designs that satisfy all of the ESBWR design requirements and have been reviewed and approved by the NRC are to be used during the initial core (Cycle 1) in any facility that adopts the ESBWR certified design.

4.2.2 Summary of Technical Information

DCD Tier 2, Revision 9, Section 4.2.1.1, describes the design basis of the ESBWR fuel assembly. The thermal-mechanical fuel design provides the following capabilities:

- Substantial fission product retention capability during all potential operational modes to comply with 10 CFR Part 20, “Standards for Protection against Radiation,” 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and 10 CFR Part 100, “Reactor Site Criteria”

- Sufficient structural integrity to prevent operational impairment of any reactor safety equipment

The fuel assembly and its components are designed to withstand the following:

- Predicted thermal, pressure, and mechanical interaction loadings occurring during startup testing, normal operation, and AOOs; infrequent events; accidents; and mechanical loads from seismic events

- Lift loads and fuel drop events predicted to occur during fuel handling

In DCD Tier 2, Revision 9, Section 4.2 describes the fuel design and Appendix 4B provides the licensing acceptance criteria for the fuel design, along with a brief description of the design evaluations. An earlier version of DCD Tier 1, Section 2.8, provided principal fuel design and performance requirements. In the final DCD, these criteria were reclassified as Tier 2* and moved to Appendix 4B.

DCD Tier 2, Revision 9, Section 4.2.1.2, describes the design basis of the ESBWR control rods. These structures are designed to have the following capabilities:

- Sufficient mechanical strength to prevent displacement of their reactivity control material
- Sufficient mechanical strength to prevent deformation that could inhibit their motion

In DCD Tier 2, Revision 9, Section 4.2 describes the control rod design and Appendix 4C provides the licensing acceptance criteria for the ESBWR control rods, along with a brief description of the design evaluations. An earlier version of DCD Tier 1, Section 2.9, provided the principal control rod design and performance requirements. In the final DCD, these criteria were reclassified as Tier 2* and moved to Appendix 4C.

4.2.3 Staff Evaluation

The regulatory criteria and the specific fuel, control rod, and core designs presented in Tier 2 will constitute an approved design that may be used for the COL first-cycle core loading without further staff review. An approved fuel design with specific design and performance requirements is a foundation for determining the acceptability of the plant systems' response to AOOs and postulated accidents.

To fulfill these regulatory requirements, the DCD references the approved GE14E fuel assembly design documented in the following licensing topical reports (LTRs): NEDC-33240P, Revision 1, "GE14E Fuel Assembly Mechanical Design Report," and NEDC-33242P Revision 2, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report." The approved Marathon control rod design is documented in topical report NEDE-33243P, Revision 2, "ESBWR Control Rod Nuclear Design Report," and NEDE-33244P, Revision 1, "ESBWR Marathon Control Rod Mechanical Design Report." The staff documented the basis for its approval of the GE14E fuel design and the Marathon control rod design in the safety evaluations for LTRs NEDC-33240P, Revision 1; NEDC-33242P, Revision 2; NEDE-33243P, Revision 2; and NEDE-33244P, Revision 1. The safety evaluation report (SER) for NEDC-33326P Revision 1, "GE14E for ESBWR Initial Core Nuclear Design Report," provides the staff evaluation of the initial core fuel design and core loading pattern.

During the July 2007 design audit of the GE-Hitachi Nuclear Energy (GEH) control rod and fuel assembly, the staff found that the mechanical design of the ESBWR Marathon control rod blade differed from that presented in NEDE-33243P and NEDE-33244P. The staff requested GEH to issue a revision to these reports that would document the revised design of the ESBWR Marathon and also capture any applicable responses to requests for additional information (RAI) from the staff's review of the Marathon-5S control rod design for use in operating reactors (RAI 4.9-12). GEH responded to RAI 4.9-12 by noting that it had addressed the differences in NEDE-33243P, Revision 2, and NEDE-33244P, Revision 1. The staff reviewed the LTRs which showed that the differences were addressed, therefore; RAI 4.9-12 is resolved.

4.2.3.1 *ESBWR DCD Tier 1*

The applicant has reclassified the ESBWR fuel and control rod principal design and performance requirements, which originally were specified in DCD Tier 1, Revision 3, Sections 2.8 and 2.9 and moved them to DCD Tier 2, Appendices 4B and 4C (respectively) (See Section 4.2.3.2 below).

Even though the applicant deleted the ITAAC for fuel and control rod design in later revisions of the DCD, the following fuel-related design commitments are included in the ITAAC in DCD Tier 1, Table 2.1.1-3 for the reactor pressure vessel (RPV) system for verification:

- The initial fuel to be loaded into the core will withstand flow-induced vibration and maintain fuel cladding integrity during operation.

- The fuel bundles and control rods for initial core have been fabricated in accordance with the approved fuel and control rod design.

- The reactor internals arrangement will conform to the fuel bundle, instrumentation, neutron sources, and control rod locations shown in DCD Tier 1, Figure 2.1.1-2.

In addition, the ITAAC for the nuclear boiler system in DCD Tier 1, Table 2.1.2-3 include the following design commitments:

- The pressure loss coefficient of each of the following components is within the uncertainty band of the pressure loss coefficient used in the natural circulation flow analysis:

  – Steam separator
  – Fuel bundle
  – Fuel support piece orifice
  – Control rod guide tubes
  – Shroud support

- The hydraulic diameter, the geometry of heated surfaces, and flow area in fuel assemblies are within the uncertainty band of the geometry used in the natural circulation flow analysis.

During the review of the GE14E fuel assembly design, the staff issued RAI 4.8-7 to request an explanation regarding the lack of mechanical testing for flow-induced vibration for the proposed bundle design. In response, GEH proposed specific flow-induced vibration testing for the design of any fuel assembly to be loaded in the ESBWR. DCD Tier 1, Section 2.1.1, Table 2.1.1-3, lists the required testing. NEDC-33240P identifies the acceptance criteria specific to the GE14E fuel design. Therefore, based on the applicant's response, RAI 4.8-7 is resolved.

4.2.3.2 *ESBWR DCD Tier 2*

The fuel system consists of the fuel assembly and the reactivity control assembly (control rod). The fuel assembly consists of the fuel bundle, channel, and channel fastener. The fuel bundle consists of full-length and part-length fuel rods (some of which may contain burnable neutron absorbers), water rods, spacers, springs, and assembly fittings DCD Tier 2, Appendix 4B contains a set of design criteria to be satisfied by new fuel designs that are to be loaded into an ESBWR.

A previous version of DCD Tier 2, Section 4.2.1.1.4, stated that the cladding oxide thickness itself is not separately limiting, and therefore, no design limit on cladding oxide thickness is specified. Likewise, a previous version of DCD Tier 2, Section 4.2.1.1.5, stated, "Mechanical properties testing demonstrates that the cladding mechanical properties are negligibly affected for hydrogen contents far in excess of that experienced during normal operation." The staff was concerned that these statements were too general and needed to be supported by mechanical testing data. The staff issued RAI 4.2-2 and RAI 4.2-4 to request that corrosion limits, expressed as oxide thickness in microns and hydrogen content in parts per million, be quantified for each fuel rod design. At a minimum, the basis of these design limits should include: (1) an oxide thickness that has been specifically accounted for in mechanical design calculations and that limits localized surface defects that may promote nonuniform mechanical properties and (2) a hydrogen content limit that maintains the cladding strain design limit (e.g., 1.0-percent plastic plus elastic strain).

After several supplements regarding the original RAI requests, GEH proposed specific corrosion limits that support the fuel mechanical design and cladding strain criterion for the GE14E fuel design. Section 3.2 of the SER for NEDC-33240P and NEDC-33242P documents the basis for NRC's approval of the corrosion limits for GE14E and therefore, RAI 4.2-2 and RAI 4.2-4 are resolved.

DCD Tier 2, Section 4.2.3.1, refers to the GSTR-Mechanical (GSTRM) Fuel Model topical report NEDC-31959P, "Fuel Rod Thermal Analysis Methodology (GSTRM)," issued April 1991, as the approved fuel rod thermal-mechanical design model. The staff issued RAI 4.2-3 requesting the licensing history of GSTRM, including the staff's review and any subsequent changes to the various fuel performance models within GSTRM. In response, the applicant provided documentation on GSTRM and identified several code modifications. In addition, the applicant updated the cited GSTRM report in the DCD. Based on the applicant's response and the documentation they provided regarding code modifications, RAI 4.2-3 is resolved.

While performing FRAPCON-3 benchmark calculations in support of the GE14E fuel assembly design topical report, the staff identified a potential nonconservatism in the GSTRM fuel temperature calculation. It is believed that the lack of a burnup-dependent uranium oxide (UO_2) thermal conductivity model is responsible for differences observed between identical FRAPCON-3 and GSTRM calculations. A nonconservative fuel temperature prediction would impact several thermal-mechanical design analyses (e.g., fuel melt, fission gas release) and subsequently, the input to safety analyses (e.g., loss-of-coolant accident [LOCA] stored energy, gap conductivity). The staff accepts the use of the GSTRM model for both gap conductance and thermal conductivity in the ESBWR design certification. The conclusions and limitations for ESBWR TRACG LOCA analyses contained in the staff evaluation of the GEH Part 21 report (Appendix F to the safety evaluation for NEDC-33173P) are applicable to this safety evaluation. The NRC must approve the use of other methods or analysis strategies for the ESBWR. Details of staff evaluation of this issue are included in Section 21.6.3.2.14 of this report.

A previous version of DCD Tier 2, Section 4.2.4.9, stated, "Subsequent Marathon designs or absorber section loadings will be within ±5 percent Δk/k of the initial ESBWR Marathon design." The staff issued RAI 4.2-9 to request clarification of the meaning and the intent of this sentence. In response regarding the proposed requirement and change criteria, GEH agreed to remove any implied change process and to revise the DCD text accordingly. The staff finds the revised text in DCD Tier 2, Section 4.2.4.9 acceptable; therefore, RAI 4.2-9 is resolved.

The summary of the changes made in the fuel topical reports were reviewed by the staff and as documented in Section 3.2 of the SER for NEDC-33240P and NEDC-33242P, the NRC approved the GE14E fuel design up to the specified rod power envelopes subject to the limitations in the NEDC-33240P and NEDC-33242P SER.

4.2.3.2.1 Appendix 4B Fuel Licensing Acceptance Criteria

The original text of Appendix 4B was modeled after GESTAR-II and appeared to be an overview of a fuel design change process. DCD Tier 2, Revision 9, Appendix 4B defines the specific Tier 2 and Tier 2* thermal and mechanical fuel design and performance requirements. A separate fuel assembly mechanical design topical report (or a COL application) will address these requirements to demonstrate, using approved models and methods, the acceptability of a proposed fuel assembly design for the ESBWR. The design certification process requires that the NRC specifically review and approve the fuel assembly design employed in the initial core (Cycle 1) in any facility that adopts the ESBWR certified design.

In response to RAI 4.2-5 regarding the documented change process, the applicant stated that it would revise Appendix 4B to remove all of the design process information and provided a significantly revised version. Staff concerns with the proposed revision included the lack of specific Tier 2 and Tier 2* fuel thermal and mechanical design requirements and the continued inclusion of a critical power correlation change process. In response to RAI 4.2-5, S01-S03, GEH defined specific thermal-mechanical design and performance requirements and removed the description referring to the change process mentioned in the preceding paragraph. DCD Tier 2, Revision 4 incorporates these changes. Based on the applicant's response, RAI 4.2-5 is resolved.

Principal Fuel Design and Performance Requirements

DCD Tier 2, Revision 9, Appendix 4B.1, states that the specific fuel design to be used in any facility that adopts the ESBWR certified design must comply with the following fuel design and performance requirements, which are based on the fuel requirements for the advanced boiling-water reactor (ABWR):

- Fuel rod failure is predicted not to occur as a result of normal operation and AOOs.
- Control rod insertion will not be prevented as a result of normal operation, AOOs, or postulated accidents.
- The number of fuel rod failures will not be underestimated for postulated accidents.
- Fuel coolability will be maintained for all design-basis events, including seismic and LOCA events.
- SAFDLs (thermal and mechanical design limits) will not be exceeded during any condition of normal operation, including the effects of AOOs.
- In the power operating ranges, the prompt inherent nuclear feedback characteristics will tend to compensate for a rapid increase in reactivity.
- The reactor core and associated coolant, control, and protection systems will be designed to ensure that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

Note that the following text and RAI responses may refer to DCD Tier 1, Section 2.8, instead of the final location of the requirements in DCD Tier 2, Appendix 4B, as Tier 2* criteria. This is because the design and performance requirements were reclassified after these RAIs were responded to by GEH (i.e., after DCD Revision 3).

DCD Tier 1, Revision 1, Section 2.8 defined six principal design requirements. In RAI 4.2-13, the staff requested clarification on whether these six requirements are Tier 1 fuel design requirements. In response, the applicant stated that the ABWR DCD Tier 1 design requirements were more appropriate than those originally defined for the ESBWR (in DCD Tier 1, Revision 1). As a result, the DCD Tier 1, fuel design requirements were modified (as shown above). DCD Tier 1, fuel design requirements (1) through (5) above conforms to the regulatory criteria specified in DCD Tier 2, Section 4.2.1. Therefore, based on the applicant's response, RAI 4.2-13 is resolved.

If fuel design requirement (6) is met, the fuel design complies with GDC 11. If fuel design requirement (7) is met, the fuel design complies with GDC 12. Based on consistency with past certified designs and compliance with current regulatory criteria, the staff finds the fuel design requirements acceptable.

Principal Fuel Channel Design and Performance Requirements

DCD Tier 2, Revision 9, Appendix 4B.1, states that the specific fuel channel design to be used in any facility that adopts the ESBWR certified design must comply with the following three principal fuel channel design requirements:

- During any design-basis events, including the mechanical loading from a safe-shutdown earthquake event combined with LOCA event, fuel channel damage should not be so severe as to prevent control rod insertion when it is required.
- Coolability will be maintained for all design-basis events.
- Channel bowing will not cause SAFDLs to be exceeded during normal operation and AOOs.

Although these requirements now reside as Tier 2* criteria in DCD Tier 2, Appendix 4B, due to reclassification after GEH responded to these RAIs, the following text and RAI responses may refer to DCD Tier 1. In RAI 4.2-13 the staff requested that the applicant provide clarification whether the fuel design requirements are in fact Tier 1 requirements, noting that the ESBWR licensing approach differs from that of the ABWR. In response, the applicant stated that the ABWR DCD Tier 1 fuel channel design requirements were more appropriate than those defined for the ESBWR in DCD Tier 1, Revision 1. GEH modified the Tier 1 fuel channel requirements to the fuel design requirements shown above. As part of this modification, the requirement “to ensure that channel deflection does not preclude control rod drive operation” was removed. Recent operating experience has demonstrated that channel bow may significantly impact control rod movement. Control rod blade-to-channel clearance, blade design and materials, and burnup history affect channel deflection and its potential impact on control rod movement. The staff finds the removal of this requirement acceptable because Tier 1 design requirements related to control rod insertion, which capture potential effects of channel bow, remain for both the fuel design and control rod design. The DCD Tier 2, Revision 3, Appendix 4B, design criteria are in agreement with the regulatory requirements and are acceptable, therefore; based on the applicant’s response, RAI 4.2-13 is resolved.

Fuel Thermal-Mechanical Design Requirements

The revised Tier 2* fuel thermal-mechanical design requirements provided by GEH in response to RAI 4.2-5 S01, are listed below:

(1) The cladding creepout rate due to fuel rod internal pressure shall not exceed the fuel pellet irradiation swelling rate.

(2) The maximum fuel center temperature shall remain below the fuel melting point.

(3) The cladding circumferential plastic strain during an AOO shall not exceed 1.00 percent.

(4) The fuel rod cladding fatigue life usage shall not exceed the material fatigue capability.

(5) Cladding structural instability, as evidenced by rapid ovality changes, shall not occur.

(6) Cladding effective stresses/strains shall not exceed the failure stress/strain.

(7) Fuel pellet evolved hydrogen at greater than 1,800 degrees Celsius (C) (3,272 degrees Fahrenheit [F]) shall not exceed prescribed limits.

With the exception of the fuel melt design limit (i.e., requirement [2]) and cladding strain design limit (i.e., requirement [3]), the revised Tier 2* fuel design requirements are consistent with currently approved fuel design criteria and are acceptable.

With respect to fuel melting, the staff had concerns about allowing limited fuel melting during an AOO and the definition of core-wide versus local events. In a previous version of DCD Tier 2, Appendix 4B.2 stated, "For local AOOs such as rod withdrawal error, a small amount of calculated fuel pellet centerline melting may occur, but is limited by the 1 percent cladding circumferential plastic strain criterion." In RAI 4.2-6 the staff expressed concerns with: (1) the ability to accurately model fuel volumetric expansion as fuel enthalpy approached incipient melt temperatures and (2) the ability to accurately model the involved fuel pellets in future operation. In response, the applicant stated that the rod withdrawal error during refueling has been classified as an infrequent event and that it would remove the statement regarding fuel pellet melting and revise the DCD accordingly. Based on the applicant's response and subsequent update to the DCD, RAI 4.2-6 is resolved. Chapter 15 of this report discusses and resolves the reclassification of Chapter 15 events, which was an open item in RAI 15.0-15.

Furthermore, the staff would not accept fuel melting for any AOO or infrequent event. On a related subject, the interim criterion for reactivity-initiated accidents (e.g., control rod drop) precludes fuel melting in order to meet the requirements of GDC 28, "Reactivity limits."

In DCD Tier 2, Revision 3, Appendix 4B Tier 2* fuel design requirements, the text states, "…fuel melting during normal steady-state operation and whole core anticipated operational occurrences are not expected to occur." This statement implies that it is acceptable to experience fuel melt during local AOOs. In Revision 6 of DCD Appendix 4B, Section 4B.2, item (2), "Fuel temperature," the above statement was revised to state, "…fuel melting, during normal steady-state operation and anticipated operational occurrences does not occur." The staff finds this acceptable because it satisfies the Section 4B criteria.

Based on the above, the staff finds the Tier 2* fuel thermal-mechanical design requirements (1) through (7) acceptable.

Nuclear Design Requirements

The revised Tier 2* nuclear design requirements are listed in DCD Appendix 4B and are listed below:

- A negative Doppler reactivity coefficient is maintained for any operating condition.

- A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels is maintained for any operating conditions.

- A negative moderator temperature reactivity coefficient is maintained for temperatures equal or greater than hot shutdown.

- To prevent a super prompt critical reactivity insertion accident originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel is negative.

- A negative power reactivity coefficient (as determined by calculating the reactivity change due to an incremental power change from a steady-state base power level) is maintained for all operating power levels above hot shutdown.

- The core is capable of being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive control rod, or rod pair, in the full-out position and all other rods fully inserted.

The six Tier 2* nuclear design requirements are consistent with those listed for the ABWR (incorporated by reference to Section 4B.4 of the ABWR DCD, Revision 4). The nuclear design requirements related to fuel storage and mixed-vendor fuel loading were removed. DCD Tier 2, Revision 9, Section 9.1 addresses the requirements regarding fuel storage criticality. Mixed-vendor fuel loading is not applicable to the initial core. Based on consistency with past certified designs and compliance with current regulatory criteria, the staff finds the Tier 2* nuclear design requirements (1) through (6) acceptable.

Nuclear design requirements (1) through (5) satisfy the requirements of GDC 11 in that the net effect of prompt inherent nuclear feedback characteristics in the core tend to compensate for rapid increases in reactivity when operating in the power range. With respect to nuclear design requirement (6), covered in RAI 4.3-10, the staff had concerns that this requirement was not specific to the CRD system and that shutdown margin requirements could be interpreted as including the standby liquid control system (SLCS). The applicant's response for RAI 4.3-10 resolved these concerns by clarifying that the SLCS is not included in the CRD requirements.

The response to RAI 4.3-10 stated that:

> "….if the selected rods were not neutronically coupled, then the worth of the hydraulic control unit (HCU) rod pair would be equal to the sum of the worth of the individual rods. The individual rod worth for each HCU pair would then be additive and one would conclude (stated in Part C of RAI 4.3-10) that SLCS is required to achieve sub-criticality with an HCU failure."

The staff agrees that HCU rods are loosely coupled, rod worth is not additive, and sufficient shutdown margin exists in the event of an HCU failure without the need for SLCS. Based on the applicant's response, the staff finds that sufficient shutdown margin exists in the case of an HCU failure. Based on the applicant's response, RAI 4.3-10 is resolved.

Critical Power Design Requirements

The design certification process requires that the NRC specifically review and approve the fuel assembly design, along with its critical power correlation, for the initial core loading (Cycle 1) in any facility that adopts the ESBWR certified design. The change process described in Section 4B.3 was not acceptable because it implied that changes to the correlation are acceptable without NRC review.

DCD Tier 2, Revision 0, Appendix 4B.5, stated, "99.9 percent of the rods in the core must be expected to avoid boiling transition for core-wide incidents of moderate frequency…." This criterion differs from GESTAR-II, which states, "Ninety nine point nine percent (99.9%) of the rods in the core must be expected to avoid boiling transition." In response to RAI 4.2-7 regarding this apparent change in philosophy, the applicant stated that it would revise the text to be consistent with GESTAR-II. In response to RAI 4.2-7 S01, the applicant decided to remove this text "because it is already covered in Chapter 15 of the DCD." Based on the applicant's response and the removal of the "moderate frequency" statement, RAI 4.2-7 is resolved.

4.2.3.2.2 Appendix 4C, Control Rod Licensing Acceptance Criteria

DCD Tier 2, Revision 0, Appendix 4C, included an overview of a control rod design change process. This appendix should have defined the specific Tier 2 and Tier 2* control rod design requirements. The staff issued RAI 4.2-8 stating that: "Revision 0 of DCD Tier 2, Section 4C.1, states, '…designs meeting the following acceptance criteria are considered to be approved and do not require specific NRC review.'" This quoted statement was inaccurate. The NRC must specifically review and approve the control rod design employed in the initial core (Cycle 1) in any facility that adopts the ESBWR certified design. The staff requested that the applicant define the specific Tier 2 and Tier 2* in the CRD requirements. In response, the applicant agreed to revise the text by removing the implied change process. The staff reviewed and accepted the revised text in DCD Tier 2, Revision 3, Appendix 4C, and based on the applicant's response, RAI 4.2-8 is resolved.

Control Rod Design Requirements

DCD Tier 1, Revision 1, Section 2.9, provided the following four principal control rod design requirements:

- The control rod stresses, strains, and cumulative fatigue will be evaluated so that they do not exceed the ultimate stress or strain limit of the material, structure, or welded connection.

- The control rod will be evaluated to be capable of insertion into the core during all modes of plant operation within limits assumed in plant analyses.

- The material of the control rod will be compatible with the reactor environment.

- The plant core analyses will include the reactivity worth of all control rods.

DCD Tier 1, Revision 1, Section 2.9, and Appendix 4C.1 include a control rod design requirement that states, "…lead surveillance control rods may be used." The staff issued RAI 4.2-10 requesting clarification because; whether in a Tier 1, Tier 2, or Tier 2* design requirement, the use of the term "may" needs to be revisited. In other words, there should always be an indication of this type or magnitude of design change if it would warrant in-reactor service before batch implementation. In response, the applicant removed any design requirements related to lead surveillance of control rods. Because of the requirement that control rod designs are NRC reviewed and approved, the staff accepted the deletion, and based on the applicant's response, RAI 4.2-10 is resolved.

DCD Tier 1, Revision 1, Section 2.9, and Appendix 4C.1 define principal design criteria for the control rod. One of the design criteria (in a previous revision) stated that the stresses, strains, and cumulative fatigue will be evaluated so that they do not exceed the ultimate stress or strain limit of the material. Certain boiling-water reactor (BWR) control rod designs include long axial welds between the square tubes and welds connecting the absorber wings to the handle and connector. The staff issued RAI 4.2-14 to request that the applicant demonstrate that the structural properties (e.g. weld regions) are never more limiting than the material properties. In their response regarding the structural properties versus material properties of the control rod, the applicant agreed to revise the design requirement to include the structure and welded connection. The applicant also described mechanical testing that demonstrates that the base material fails before any of the welds. Based on the applicant's response, the staff finds design requirement (1) above acceptable; therefore RAI 4.2-14 is resolved.

The discussion of principal design criteria in DCD Tier 1, Revision 1, Section 2.9, states, "The material of the control rod will be compatible with the reactor environment." In RAI 4.2-11 the staff noted in recent years the phenomena of shadow corrosion has been identified. Those phenomena are partly due to the interaction between the Zircaloy channels and stainless steel control rods. The staff requested that the applicant discuss the implementation of this design criterion with respect to shadow corrosion. In response, the applicant stated that this design requirement was related to stress-corrosion cracking (SCC) resistance of the material and deformation induced by B_4C swelling. The applicant also discussed shadow corrosion, its effect on channel bow, and the applicant's strategy for mitigating the effects of shadow corrosion. Based on the applicant's response, the staff finds that design requirement (3) is fulfilled; therefore, RAI 4.2-11 is resolved.

Design requirements (2) and (4), related to control rod insertion and worth, are consistent with the regulatory criteria and are acceptable.

Initially, GEH included the fuel and control rod design requirements as Tier 1; later, GEH incorporated them in the DCD as Tier 2* material.

The revised control rod design requirements in DCD Tier 2, Revision 9, Section 4C.1 are listed below:

- Control rod stresses, strains, and cumulative fatigue are evaluated not to exceed the ultimate stress or strain limit of the material, structure, or the welded connection.

- The control rod design is evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.

- Control rod materials are shown to be compatible with the reactor environment.

- Control rod reactivity worth is included in the plant core analyses.

4.2.4 Conclusions

Based on the discussion above, the staff finds that the DCD Tier 2 and Tier 2* criteria related to fuel system design and performance requirements (including the control assembly design) satisfy all of the regulatory requirements and SRP guidelines identified in Section 4.2.1, including the requirements of 10 CFR 50.46; GDC 10, 27, and 35; and 10 CFR 52.47(a).

As identified in Section 4.2.1 of this report, the specific fuel, control rod, and core designs referenced within the DCD will constitute an approved design that may be used for the COL first-cycle core loading without further staff review. To fulfill these regulatory requirements, the DCD references the following NRC-approved topical reports:

- NEDC-33240P and NEDC-33242P, concerning the GE14E fuel assembly design
- NEDE-33243P and NEDE-33244P, concerning the Marathon control rod design
- NEDC-33326P, concerning the GE14E initial core nuclear design

The staff has confirmed that the cited GE14E fuel assembly design and Marathon control rod design satisfy the design and performance requirements specified in DCD Tier 2, Revision 9, Appendices 4B and 4C. The staff's approval of the GE14E fuel assembly design and Marathon control rod design includes limitations and conditions, which are addressed in the safety evaluations for LTRs NEDC-33240P/NEDC-33242P and NEDE-33243P/NEDE-33244P. As such, the staff finds the use of this fuel design system acceptable for ESBWR Cycle 1.

4.3 Nuclear Design

4.3.1 Regulatory Criteria

DCD Tier 2, Revision 9, Section 4.3.1, presents the ESBWR nuclear design bases. The staff reviewed DCD Tier 2, Revision 9, Section 4.3.1 in accordance with the regulatory guidance for the review of nuclear design, including adherence to applicable general design criteria (GDC) discussed in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (hereafter referred to as the SRP), Section 4.3, Draft Revision 3, issued June 1996. The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any generic issues (GI), bulletins (BL), generic letters (GL), or technically significant acceptance criteria (except Appendix 4B, Interim Criteria and Guidance for the reactivity initiated accidents) beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 3 of SRP Section 4.3, issued in June 1996, is acceptable for this review.

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 reactor coolant pressure boundary or impair the capability to cool the core, or sustain unstable core conditions. To meet these objectives, the nuclear design must conform to the following GDCs:

- GDC 10, requiring the reactor design (reactor core, reactor coolant system, control and protection systems) are designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs

- GDC 11, requiring a net prompt inherent negative feedback power coefficient in the operating range

- GDC 12, requiring that power oscillations that can result in conditions exceeding SAFDLs are not possible, or can be reliably and readily detected and suppressed

- GDC 13, "Instrumentation and control," requiring a control and monitoring system to monitor variables and systems to assure adequate safety including those that can affect the fission process over their anticipated ranges for normal operation, AOOs, and accident conditions

- GDC 20, "Protection system functions," requiring, in part, a protection system that automatically initiates a rapid control rod insertion to ensure that fuel design limits are not exceeded as a result of AOOs

- GDC 25, “Protection system requirements for reactivity control malfunctions,” requiring protection systems designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems

- GDC 26, “Reactivity control system redundancy and capability,” requiring, in part, two independent reactivity control systems of different design principles that are capable of holding the reactor subcritical under cold conditions

- GDC 27 requiring, in part, a control system designed to control reactivity changes during accident conditions in conjunction with poison addition by the emergency core cooling system (ECCS)

- GDC 28, requiring, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding

The acceptance criteria in the area of nuclear design, specifically power distributions, are based on meeting the relevant requirements of the GDCs related to the reactor core and the reactivity control systems.

The nuclear design basis for control requirements is that SAFDLs are met during normal operation and AOOs. Therefore, the maximum linear heat generation rate (MLHGR) and the minimum critical power ratio (MCPR) constraints shall be met during operation. The MLHGR limit and operating limit MCPR (OLMCPR) are determined such that the fuel rods do not exceed licensing limits during AOOs.

The MLHGR is the maximum local linear heat generation rate (LHGR) (more specifically, that of the fuel rod with the highest surface heat flux at any nodal plane in a fuel bundle in the core). The MLHGR operating limit is bundle-type dependent, and LTR NEDC-33242Pdescribes the determination of this limit. The LHGR is monitored to ensure that all mechanical design requirements are met. The fuel will not be permitted to be operated at LHGR values greater than those found to be acceptable within the body of the safety analysis under normal operating conditions. Under abnormal conditions, including the maximum overpower condition, the MLHGR will not cause fuel melting or cause the strain limit to be exceeded.

The MCPR is the minimum critical power ratio of all of the fuel bundles. The critical power ratio (CPR) for any bundle is the ratio of the bundle power that would result in transition boiling to the current bundle power. Therefore, the bundle with the smallest CPR has the smallest margin to transition boiling. The CPR is a function of several parameters; the most important are bundle power, bundle flow, the local power distribution, and the details of the bundle mechanical design.

The plant OLMCPR is established by considering the limiting AOOs for each operating cycle. The OLMCPR is determined such that 99.9 percent of the rods avoid boiling transition during the limiting analyzed AOO, as discussed in LTR NEDC-33237P, Revision 4 “GE for ESBWR - Critical Power Correlation, Uncertainty, and OLMCPR Development,“ July 2008.

To meet the provisions of GDC 10, the design bases affecting power distribution of the ESBWR include the following parameters:

- Under abnormal conditions (including maximum overpower), the MLHGR will not cause the fuel to exceed mechanical design limits.

- The MCPR during normal operation will remain greater than the OLMCPR to avoid boiling transition during normal operation and AOOs.

GDC 13 provides the required criteria to evaluate core monitoring. Core monitoring is performed using in-core nuclear instrumentation, in part to ensure that the core is being operated within these limits and to ensure that automatic reactivity control systems are initiated during adverse plant transients so that SAFDLs are met.

GDC 20, 25, 26, and 27 provide the required criteria for the reactivity control system. The control rod system is designed to provide shutdown margin and reactivity control of maximum excess reactivity anticipated during cycle operation. The control rods provide reactivity changes that compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load and allow for control of the power distribution within the core.

GDC 12 specifies the requirements relative to reactor stability. The staff has documented its review of the compliance of the ESBWR with the provisions of GDC 12 in Section 4A of this report.

The staff separately reviewed the compliance of a proposed initial core design that was submitted in LTR NEDC-33326P, Revision 1. The staff's review of the initial core nuclear design, in accordance with the aforementioned review criteria, is documented separately in the staff safety evaluation of NEDC-33326P.

4.3.2 Summary of Technical Information

Core Description

The 4,500-megawatt-thermal ESBWR core consists of 1,132 fuel bundles and 269 control rods. Several types of fuel bundles, similar except for differences in enrichment and burnable poison content, are loaded in the reference pattern. The purpose of the bundle differences is to allow for a flatter radial power distribution across the core and provide low reactivity assemblies that are similar in their neutronic behavior to partially burnt assemblies.

Core Monitoring

The ESBWR core monitoring is accomplished with several in-core nuclear instruments that cover the expected ranges for normal operation, AOOs, and accident conditions. The neutron monitoring system comprises three separate measurement systems: the source range neutron monitor (SRNM), the local power range monitor (LPRM), and the automatic fixed in-core probe (AFIP). The power range neutron monitoring system (PRNM) receives signals from several local detectors. These in-core nuclear instruments include the LPRMs, as well as automatic fixed in-core gamma thermometers (GTs). For low powers characteristic of the source range through a normal startup (greater than 10 percent of rated thermal power), the core neutron flux is monitored using the SRNM system.

The LPRMs are arranged in 64 strings, each with four detectors, and distributed throughout the core. DCD Tier 2, Revision 9, Figure 7.2-7, shows the locations of LPRM strings. For every four-by-four array of bundles, there are four LPRM strings (one at each corner). The LPRM strings comprise four LPRM detectors that are spaced evenly axially throughout the core. The LPRM detectors are polarized fission chambers.

Inside the LPRM instrument guide tube are seven AFIPs. The AFIP is a gamma thermometer (GT) instrument that is used to periodically calibrate the LPRM signal. DCD Tier 2, Revision 9, Figure 7.2-8, shows the axial elevation of the AFIPs. Each LPRM instrument string contains seven AFIPs. One AFIP is at the same elevation as the midplane of each of the LPRM detectors. In between each LPRM detector, there is another AFIP. The AFIPs are evenly distributed between the uppermost and bottommost LPRMs at 381-millimeter (15-inch) intervals.

To cover the entire range of normal operation, instruments are included to measure the neutron flux and monitor the fission process in the startup range. Increased instrument sensitivity is necessary to monitor the startup process when the reactor power is very low. According to DCD Tier 2, Revision 9, Section 7.2.2.2.4.1, the SRNM comprises 12 detectors. These detectors are fixed in-core regenerative fission chamber sensors. The 12 detectors are spaced evenly throughout the core and located at the core midplane axially; DCD Tier 2, Revision 9, Figure 7.2-6, shows the radial locations. The detectors are housed within pressure barrier tubes. The SRNM detectors are capable of measuring the reactor flux over ten decades, from a flux level of approximately 10^3 neutrons per square centimeter per second ($n/cm^2/s$) to 10^{13} $n/cm^2/s$. This range extends to approximately 10 percent of rated power. The LPRM monitoring capability overlaps this range, as the LPRMs can monitor core power from the startup range through the power range, from 1 percent of power to greater than rated thermal power.

The rod control and information system (RC&IS) is nonsafety-related. The RC&IS is a logic system that provides controls on reactor maneuvering through control rod motion during normal operation and maintains status information regarding the current control rod configuration for the reactor.

Using local power indications from the LPRM detectors, the RC&IS subsystems issue rod blocks to ensure that safety and operating limits are not exceeded as a result of control rod motion. The automated thermal limit monitor and multichannel rod block monitor (MRBM) work together above the low power setpoint to ensure that rod withdrawals are inhibited when local detectors indicate power changes that challenge the MLHGR limit or the OLMCPR. The MRBM, unlike conventional rod block monitors, uses several channels of LPRM indications throughout the core to simultaneously monitor each region of the core where control rods are being withdrawn during ganged withdrawal sequences. Below the low-power setpoint, the rod worth minimizer (RWM) is used to compare the control rod withdrawal sequence at low power to a preprogrammed control rod withdrawal pattern. In cases where the control rods are withdrawn in a different manner, the RWM enforces control rod insertions and withdrawals at low power to reduce the available reactivity worth of a control rod and thus mitigate the consequences of a control rod drop accident during low-power operation.

Upon receipt of a scram signal by the reactor protection system (RPS), the RC&IS initiates a fast fine-motion control rod drive (FMCRD) run-in as a backup to the hydraulic scram through the diverse protection system (DPS). The RC&IS also sends selected control rod run-in (SCRRI) signals to the DPS following specific AOOs, namely load rejection, turbine trip, and loss of feedwater heating.

Another important function of the RC&IS is to interface with the plant computer to perform LPRM calibration and plant simulator adaptation. This function is performed by using AFIP signals in conjunction with three-dimensional nuclear models to determine gain adjustments and nodal parameter corrections.

Reactivity Coefficients

The reactivity coefficients express the effects of changes in the core conditions, such as power and fuel and moderator temperature and moderator density, on core reactivity. These coefficients vary with fuel exposure and power level.

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in predicting the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest are the Doppler coefficient, the void reactivity coefficient, and the moderator temperature coefficient. The combination of these reactivity coefficients dictates the power reactivity coefficient. A combination of negative coefficients ensures that the reactor will have an inherent negative reactivity feedback with increasing power.

To demonstrate that the Doppler reactivity coefficient remains negative in the power operating range, the applicant calculated temperature-dependent eigenvalues for each of the fuel bundle types. The Doppler reactivity coefficient is predominantly driven by the uranium-238 and plutonium-240 content in the fuel, and while an inherent feature of the fuel, this coefficient does not vary significantly between BWR fuel designs. The Doppler coefficient calculated for the ESBWR initial core is negative for increasing fuel temperature and similar in magnitude to operating reactor Doppler coefficients.

The void reactivity coefficient was estimated for both the power range of operation and for cold shutdown conditions. The applicant's analyses indicate a negative trend of core eigenvalue with increasing core average void content in the power range of operation, indicating inherent negative reactivity feedback under these conditions.

In RAI 4.3-6 the staff requested verification that the calculated values of the void reactivity coefficient at the beginning of cycle (BOC), middle of cycle (MOC) and end of cycle (EOC) at nominal operating conditions are negative. In response, the applicant provided the BOC, MOC, and EOC void reactivity coefficients predicted by PANACEA based on enthalpy perturbations to the core model. The staff finds that the results indicate a consistently large, negative void reactivity coefficient.

The magnitude of the void reactivity coefficient, however, decreases with decreasing void content. Therefore the applicant identified the cold shutdown condition as a limiting case, particularly the EOC following depletion of burnable poisons. The analysis for the limiting condition verifies that the void reactivity coefficient is negative.

Lastly, the applicant calculated the moderator temperature coefficient. During normal operation the coolant is subcooled only near the core inlet and remains at a near constant temperature once reaching saturated conditions. The EOC for the reference core loading was identified as the condition with the least negative moderator temperature coefficient. The results indicate that, at temperatures above 150 degrees C (approximately 300 degrees F), the core eigenvalue decreases with increasing water temperature.

The moderator temperature coefficient decreases in magnitude over cycle exposure with the withdrawal of control rods and the depletion of gadolinia burnable poisons. Late in the cycle, the reduction in the poison content leads to conditions where the reactor could become over moderated, thereby yielding a positive moderator temperature coefficient for cold conditions. While the EOC moderator temperature coefficient is positive, it is small compared to the effects of the void reactivity feedback. The applicant's calculations show that the moderator temperature coefficient at the EOC under cold conditions may be positive.

At cold conditions towards the EOC, the ESBWR neutron spectrum is slightly over moderated, yielding a slightly positive moderator temperature coefficient for cold conditions at the EOC. In RAI 4.3-5, the staff requested additional information regarding the moderator temperature coefficient that is slightly positive at low temperatures and EOC. (The moderator temperature coefficient remains negative for all operating conditions at and above hot standby.) In their response, the applicant stated that the moderator temperature coefficient may become positive when the reactor coolant is below rated pressure and temperature, but during these conditions, a positive moderator temperature coefficient is manageable. Below rated conditions, the reactor power is low, and therefore, the time it takes to heat the volume of coolant to result in an appreciable increase in temperature is very long. In addition, the cooling rate for the fuel would be slow if a power increase occurred, particularly since the heatup adds negative reactivity through the Doppler Effect. Based on the applicant's responses, the staff concludes that for all operating conditions (with temperatures above hot standby) the moderator temperature reactivity coefficient is negative. At temperatures below hot standby the Doppler reactivity coefficient provides a prompt feed back to counter power increases. Therefore; based on the aforementioned responses RAI 4.3-5 and RAI 4.3-6 are resolved.

Reactivity Control Systems

The control rod system is designed to provide shutdown margin and reactivity control of maximum excess reactivity anticipated during cycle operation. The control rods provide reactivity changes that compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load and allow for control of the power distribution within the core.

In addition to providing the means for controlling core reactivity for power maneuvering, the control rods provide the minimum shutdown margin following any AOO and are capable of making the core subcritical rapidly enough to prevent exceeding SAFDLs. The control rods are automatically hydraulically inserted upon receipt of a scram signal from the RPS.

The applicant has provided an analysis in DCD Tier 2, Revision 9, which shows that the control rod worth is sufficient to ensure a subcritical configuration for xenon-free, cold shutdown conditions at BOC. The BOC condition is limiting in terms of available shutdown margin.

The control rods are backed-up by the standby liquid control system (SLCS). The SLCS is a second reactivity control system meant to provide a diverse and redundant capability to the control rods. The SLCS is an accumulator-driven boron injection system. It is designed to provide the capability of bringing the reactor, at any time in a cycle, from full power with a minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state if the control rods fail to insert.

DCD Tier 2, Revision 9, Section 4.3, provides analyses of the shutdown capability of both the control rod system and the SLCS. The analyses show that either system is capable of holding the reactor subcritical at the limiting conditions in terms of exposure, temperature, and xenon. In the case of the control rod system, the calculations consider a single failure of a rod to insert and the single failure of a HCU to insert a pair of rods.

4.3.3 Staff Evaluation

Core Monitoring

The neutron monitoring system is designed to meet the requirements of GDC 13 and GDC 10. Specifically, the PRNM and SRNM are designed to monitor the fission process during normal operation and over the range of anticipated operation and accident conditions. The PRNM comprises several LPRM detectors with the capability of monitoring the neutron flux in the reactor between 1 percent of rated core power and well over 100 percent of the rated core power (125 percent). The SRNM is designed to monitor the neutron flux at very low levels (approximately 10^3 n/cm^2/s) or approximately 10 decades below the normal operating level. The combination of these two neutron monitoring subsystems allows for an overlapping monitoring capability over the full range of neutron flux levels under normal operation, including startup and AOOs. The LPRM capability extends to higher neutron flux levels, which allows for monitoring of the reactor core power during accident conditions and anticipated transients without scram (ATWS). Therefore, the staff finds that the ESBWR neutron monitoring system is acceptable in that it provides sufficient capability to adequately monitor the neutron flux levels in the reactor over the necessary ranges.

The in-core ESBWR neutron monitoring system is based on a series of distributed LPRMs. The polarized fission gas chambers are substantially the same as those instruments widely applied within the operating fleet of BWRs. The design differences between the ESBWR and conventional BWRs will not impact the fundamental operation of the LPRMs so long as the steady-state bypass void fraction remains below 5 percent as described in NEDC-33239P, Revision 4, “GE14 for ESBWR Nuclear Design Report.”

These instruments also interface with the 3D MONICORE system to determine the operating characteristics of the core. For the 3D MONICORE system to accurately assess the thermal margin during operations and to ensure that the RPS accurately detects adverse transient or accident conditions and initiates automatic protective actions such as scram, the instruments must be periodically calibrated.

The neutron monitoring system includes in-core GTs to replace the function of the traversing in-core probe system for conventional reactors. The GTs, much like gamma traversing in-core probe instruments, are used to determine the axial power shape and LPRM gain adjustment factors based on local gamma flux indications. The primary difference between the instruments is that the GTs are distributed, stationary probes.

The staff reviewed the information provided by the applicant concerning the GT design and found that, with regular calibration, the GT can be used to determine the local gamma flux. When combined with coupled transport calculations to determine the detector response kernels (or signal to power ratios), the GT indication may be used to adequately determine the local nodal power in surrounding nodes. NEDO-33197, Revision 2, “Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring,” describes the NRC-approved methodology for translating the GT signals to power distribution information. The GT instruments are spaced within the core beside the LPRMs, giving a complete radial mapping capability if the core power distribution is quadrant symmetric.

The 3D MONICORE system determines the margin to limits based on input from the neutron monitoring system, and input from the core thermal hydraulic instrumentation (i.e., core flow). The 3D MONICORE system is based on the PANAC11 calculational engine. NEDC-33239P-A, Revision 5, describes the NRC-approved PANAC11 methodology.

However, GDC 13 also requires that appropriate controls be in place to ensure that the reactor core is operated within prescribed safety and operating limits. The GDC 13 requirements for the Neutron Monitoring System are fulfilled by prescribing limits that account for instrument and measurement uncertainties. Of key importance to the prescription of these limits is the accuracy of the neutron flux measurements. The pedigree of LPRM measurements in particular is related to the efficacy of the AFIPs and process computer to effectively and accurately calibrate the local indications of the neutron flux level. The staff issued RAI 4.2-12 and RAI 4.3-2 to request additional information regarding the determination of the MLHGR value and the uncertainties in the nuclear instrumentation, calibration, biases, and the 3D MONICORE PANAC11 calculations. The staff reviewed the responses to RAI 4.2-12 and RAI 4.3-2, and the results of the review and the approval of these uncertainties are in the safety evaluation for NEDC-33239P-A and NEDE-33197P-A. (NEDC-33239P-A is the GE14 ESBWR Nuclear Design Topical Report and NEDE-33197P-A is the Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring) The uncertainties were correctly evaluated and properly applied to the operating limits. The in-core instrumentation meets the requirements of GDC 13 by providing monitoring capability over the range of expected operation and providing sufficient information, given the capabilities of the 3D MONICORE system, to monitor core operating parameters relative to associated operating limits. Therefore, based on the applicant’s responses, RAI 4.2-12 and RAI 4.3-2 are resolved.

Maintaining the reactor within the OLMCPR and operating MLHGR limit ensures that the SAFDLs are not exceeded during normal operation or as a result of AOOs. Therefore, the staff finds that the design basis satisfies GDC 10.

In summary, the staff finds that the ESBWR design adequately meets the requirements of GDC 10 and GDC 13 and is therefore acceptable.

Reactivity Coefficients

As described above, GDC 11 requires that the core be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

The applicant provided several analyses to indicate the nature and magnitude of the reactivity feedback coefficients for the reference ESBWR core. NEDC-33239P-A describes the NRC approved nuclear methods used. In each case, the applicant performed the analysis by perturbing the steady-state calculation to determine the change in eigenvalue as a result of a change in the fuel temperature, coolant temperature, or coolant void.

In general, the Doppler coefficient is a strong function of fertile heavy metal content (e.g., uranium-238) and spectrum hardness. For the ESBWR, the enrichment and planar fuel geometry are similar to operating BWRs. However, the bundle pitch is slightly greater for the ESBWR compared to operating BWRs, which leads to a softer neutron spectrum arising from increased moderation in the core bypass. The softer spectrum reduces the resonance integral and, consequently, the Doppler coefficient. The applicant's calculations are consistent with this expectation.

The increased assembly spacing also affects the moderator temperature coefficient. The increased hydrogen to heavy metal ratio decreases the magnitude of the moderator temperature coefficient and leads to slightly positive values for cold conditions at EOC where the neutron spectrum is very soft (thus, over moderated). The positive nature of the moderator temperature coefficient is of minor concern because of the relatively slow nature of the moderator temperature change (relative to fuel temperature change), and at normal operating conditions, the core dynamic behavior is driven predominantly by the strong, negative void reactivity feedback. This condition of slightly positive moderator temperature coefficient vaules is only for low temperatures and is not of sufficient magnitude to cause operational concerns during startup and shutdown operations, or a reactivity insertion problem.

The applicant provided a series of core calculations to determine the estimated void coefficient. As the void reactivity coefficient is stronger for higher void fractions, the applicant performed calculations for cold shutdown conditions. This calculation is conservative because the spectrum at cold shutdown conditions is over moderated. The applicant simulated the effects of voids in the subcooled coolant using the NRC approved PANAC11 method and found that, in the most limiting case, the void reactivity coefficient was negative.

The power reactivity coefficient is a combination of the Doppler, void, and moderator temperature reactivity coefficients. While the design differences of the ESBWR make the moderator temperature and Doppler coefficients less negative than for an operating BWR, the increased void, higher enrichment, and higher burnable poison loading result in an overall negative power coefficient. In the case of the ESBWR, the void coefficient is not significantly different from operating reactors and is a dominant contributor to the power coefficient. The staff finds the reactivity coefficient values to have been evaluated using NRC approved methods, to be negative, and ensure a negative power reactivity coefficient, therefore, they meet the requirements of GDC 11 and are acceptable.

Reactivity Control Systems

As described above, GDC 20, 25, 26, 27, and 28 specify the requirements for the reactivity control systems. The reactivity control worth calculations were performed using the TGBLA06 and PANAC11 codes. The applicant calculated the shutdown margin at several exposure points during the cycle to demonstrate that BOC is the limiting condition. The analysis provided ensures that the reactor remains subcritical with sufficient margin when the strongest rod and strongest rod pair are fully withdrawn.

On this basis, the staff finds that the control system has adequate negative reactivity worth to ensure shutdown capability, assuming that the most reactive control rod is stuck in the fully withdrawn position.

The control rod system automatically inserts control rods to shut down the reactor on receipt of a scram signal. The negative reactivity worth of the control rods is sufficient to bring the reactor to a cold-shutdown condition at any point during exposure. The core monitoring system provides operating margin to the SAFDLs. The staff finds that the ESBWR appropriately monitors the core conditions to ensure that the effects of transients do not challenge the

SAFDLs and prompts automatic scram during adverse conditions. Therefore, the design meets the requirements of GDC 20.

Additionally, the applicant explains that control rod assignments to particular HCUs shall maintain sufficient distance between rods such that there is essentially no neutronic coupling between the control cells and no significant impact on the shutdown margin given a failure of a single HCU. When the reactor is shut down, the core is filled with liquid water and the mean free paths for neutrons are much smaller than at power, where the presence of voids allows for increased neutron transport during slowing down of the neutrons. Therefore, control cell neutronic coupling is effectively limited to nearby neighboring control cells. The assignment of control rods to HCUs, such that no HCU drives two nearby control rods would preclude neutronic coupling. Without any coupling, there is no synergistic effect of a dual control rod insertion failure, which could result in local criticality.   Control rods assigned to an individual HCU are separated by several rod locations (between five and seven rod locations). As the mean free path for even higher energy neutrons at normal operating conditions ranges on the order of 15-30 cm (9.8 to 11.8 inches [in.]), and the mean free path is greatly reduced when the core is under cold conditions with control rods inserted, the staff finds that the HCU assignments adequately preclude the possibility of synergistic reactivity effects. Therefore, local criticality based on the failure of any particular HCU is not a concern if the remainder of the control rods inserted provide sufficient negative reactivity to ensure that the reactor is shutdown and subcritical under cold conditions at its most reactive point.

The staff therefore finds that the shutdown margin is sufficiently large to provide reasonable assurance that the requirements of GDC 25 are met considering the failure of a single rod or rod pair to insert.

In DCD Tier 2, Revision 9, Section 4.3.1.2 and Appendix 4B state that compliance with GDC 26 is partially demonstrated by showing margin to criticality in the most reactive cold condition with the strongest rod pair withdrawn. The staff has evaluated the calculation of the shutdown margin and reactivity margin to criticality at cold conditions assuming the strongest rod pair is withdrawn. The staff finds that the shutdown margin calculations provide reasonable assurance that the control rod system is capable of holding the reactor subcritical under cold conditions; thus, the requirements of GDC 26 are met.

The SLCS meets the requirements for diverse and redundant control systems given in GDC 26 and the combined reactivity control system requirements given in GDC 27. The staff has determined that the SLCS is adequate for bringing the reactor to a cold shutdown condition at any point in exposure and therefore acts as a fully redundant, diverse, and adequate control system. The system is diverse in that it is a dissolved poison, passive liquid injection system, thereby satisfying GDC 26. As the SLCS is fully capable of controlling the reactivity and is an ECCS, it provides sufficient negative worth to compensate for a partial failure of the control rod system, thereby satisfying GDC 27. The analysis indicates a large reactivity margin.

Analysis of the consequences of a postulated control rod drop accident (CRDA) demonstrates compliance with GDC 28. The staff reviewed the methodology and finds it to be appropriate for the design certification analysis. The staff issued RAI 4.6-23 S02 to request that the applicant demonstrate compliance with GDC 28 regarding pressure boundary integrity and acceptable radiological consequences in case of a control rod drop accident (CRDA). In response, GEH stated that the most reactive rod is assumed to get separated from the drive mechanism, get caught, and then drop to where the rod mechanism is. This scenario literally satisfies the provisions of GDC 28. The analyses accounted for the rod reactivity, fuel burnup, and cladding hydrogen content and calculated a conservative value of the fuel enthalpy. The results showed that the enthalpy rise is within the limits of the curves in SRP Section 4.2, Appendix B, Revision 3. Therefore, the design meets the requirements of GDC 28, and based on the applicant's response, RAI 4.6-23 S02 is resolved.

On the basis of its review, the staff concludes that the functional design of the ESBWR reactivity control systems meets the requirements of GDC 20, 25, 26, 27, and 28, and therefore, is acceptable. The staff separately reviewed and verified compliance of the proposed initial core design that was submitted as an LTR (NEDC-33326P). The staff safety evaluation for NEDC-33326P separately documents the staff's review of the initial core nuclear design, in accordance with the aforementioned review criteria.

4.3.4 Conclusions

The applicant described the computer programs and calculation techniques used to predict the nuclear characteristics of the reactor design. The applicant has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the ESBWR.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, significant excess reactivity is designed into the core. The applicant provided substantial information related to core reactivity requirements for the equilibrium cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor at any time during the cycle, with the highest worth control rod HCU stuck in the fully withdrawn position.

On the basis of its review, the staff concludes that the applicant's assessment of reactivity control requirements over the equilibrium core cycle is suitably conservative, and that the control system provides adequate negative worth to ensure shutdown capability.

The staff concludes that the nuclear design is acceptable and meets the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. This conclusion is based on the following:

- The applicant meets the requirements of GDC 11 with respect to inherent negative nuclear feedback characteristics in the power operating range by calculating a negative power coefficient of reactivity and using calculation methods that have been found acceptable.

- GDC 12 specifies the requirements related to reactor stability. Section 4A of this report documents the staff review of the compliance of the ESBWR with the provisions of GDC 12.

- The applicant meets the requirements of GDC 13 by providing instrumentation and controls to monitor variables and systems that can affect the fission process by providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables (such as temperature and pressure), and by providing suitable alarms and/or control room indications for these monitored variables.

- The applicant meets the requirements of GDC 26 by providing two independent reactivity control systems of different designs by having a system than can reliably control AOOs, having a system that can hold the core subcritical under cold conditions, and having a system that can control planned, normal power changes.

- The applicant meets the requirements of GDC 27 (i.e., with respect to reactivity control systems) by having a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions. This is accomplished by providing a movable control rod system and a liquid poison system, and by performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth stuck rod.

- The applicant meets the requirements of GDC 28 (i.e., with respect to postulated reactivity accidents) by demonstrating that the consequences of a postulated CRDA are sufficiently benign that the limits specified in SRP Section 4.2 are not challenged.

- The applicant meets the requirements of GDC 10, 20, and 25 with respect to SAFDLs by providing analyses demonstrating that normal operation, including the effects of AOOs, meet fuel design criteria; that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs and ensures the automatic operation of systems and components important to safety under accident conditions; and that no single malfunction of the reactivity control system causes violation of the fuel design limits.

4.4 Thermal and Hydraulic Design

In review of the ESBWR thermal-hydraulic design, the staff considered information contained in the DCD, responses to the staff's RAIs, and the topical reports referenced by the applicant. In addition, the staff conducted its review in accordance with the guidelines provided by SRP Section 4.4, Revision 2. As described in the following sections, the thermal-hydraulic design of the reactor core provides adequate heat transfer compatible with the heat generation distribution in the core.

4.4.1 Regulatory Criteria

DCD Tier 2, Revision 9, Section 4.4, presents the ESBWR thermal-hydraulic design bases and functional requirements of the fuel, core, and reactivity control system. The staff reviewed DCD Tier 2, Revision 9, Section 4.4 in accordance with the regulatory guidance for the review of thermal hydraulic design, including adherence to applicable general design criteria (GDC) discussed in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (hereafter referred to as the SRP), Section 4.4, Draft Revision 2, issued June 1996. The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any generic issues (GI), bulletins (BL), generic letters (GL), or technically significant acceptance criteria (except Appendix 4B, Interim Criteria and Guidance for the reactivity initiated accidents) beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 2 of SRP Section 4.4, issued in June 1996, is acceptable for this review.

The principal thermal-hydraulic design basis for the ESBWR reactor core is to ensure adequate heat removal to prevent fuel damage during any condition of normal operation, including the effects of AOOs. GDC 10 specifies that the reactor core and 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. Section 4.3

and DCD Tier 2, Revision 9, Appendix 4D discuss the thermal-hydraulic stability performance of the reactor. Sections 4.3, 4.A, and 21.6 of this report address the requirements of GDC 12.

Acceptance criteria are based on the following GDC:

- GDC 10, as it relates to the reactor core and associated coolant, control, and protection systems being designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs, and

- GDC 12, as it relates to the reactor core and associated coolant, control, and protection systems being designed to ensure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed

Additionally, the staff considered the regulatory guidance in the following documents:

- SRP Section 14.3.4, issued March 2007

- Three Mile Island (TMI) Action Item II.F.2 of NUREG–0737, “Clarification of TMI Action Plan Requirements,” issued November 1980, for instrumentation provided for indication of inadequate core cooling

- Design description and proposed procedures for use of the loose parts monitoring system (LPMS), consistent with the guidance of Regulatory Guide (RG) 1.133, Revision 1, “Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors,” issued May 1981 (see Section 4..4.2.11 of this report) and

- Preoperational and initial startup test program recommendations of RG 1.68, Revision 3, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” issued March 2007

The staff’s review covered the thermal-hydraulic design of the core and reactor coolant system to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, and (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs. The review also assessed the methods used to determine hydraulic loads on the core and reactor coolant system components during normal operation and design-basis accident conditions. Section 3.9.5 of this report discusses component structural evaluation.

SRP Section 4.4, Revision 3, contains the specific review criteria used by the staff in its review. The staff review included the portions of DCD Tier 1, Section 2.1.1 and Section 2.1.2 related to thermal-hydraulics. The Tier 1 design information submitted by the applicant includes the top-level design features and performance standards that pertain to the safety of the plant and include descriptive text and supporting figures. The Tier 1 information has been derived from Tier 2. The staff also evaluated the core safety limits and their respective bases. These appear in DCD Tier 2, Revision 9, Chapter 16 and Section 16B.

4.4.2 Summary of Technical Information

The ESBWR design is similar to that of the operating BWRs, except that the recirculation pumps and associated piping are eliminated. Circulation of the reactor coolant through the ESBWR core is accomplished via natural circulation. The natural circulation flow rate depends on the difference in water density between the downcomer region and the core region. The core flow varies according to the power level, because the density difference varies with changes in power levels.

To optimize flow with minimal resistance, fuel assemblies for the ESBWR design are shorter than those of operating BWRs by approximately 0.6 meters (m) (2 feet [ft]). Because of this, grid spacer separation and part-length rod height vary from those of conventional BWR fuel assemblies, resulting in differing flow patterns within the fuel bundles.

ESBWR DCD Tier 2, Revision 9, Section 4.4, describes the ESBWR design bases and functional requirements used in the thermal-hydraulic design of the fuel, core, and reactivity control system and relate these design bases to the applicable GDC. Thermal-hydraulic design of the core shall establish the thermal-hydraulic safety limits for use in evaluating the safety margin in accordance with GDC 10. The objective for normal operation and AOOs is to maintain nucleate boiling and thus avoid a transition to film boiling. Limits are specified to maintain adequate margin to the onset of the boiling transition. The key parameter used for plant operation is the CPR, or the ratio of the bundle power at which some point within the assembly experiences onset of boiling transition to the operating bundle power. Thermal margin is stated in terms of the MCPR that corresponds to the most limiting fuel assembly in the core.

DCD Tier 2, Revision 9, Section 4.4, references NEDC-33237P for discussion of the development and application of the General Electric Critical Quality Boiling Length (GEXL)14 critical power correlation for the ESBWR GE14E fuel. The GEXL14 correlation has been used for evaluation of the commercially available GE14 fuel, a conventional 12-foot long, 10x10 fuel bundle design. The shortened ESBWR (GE14E) fuel assemblies will use components identical to those used in the GE14 fuel design. These include lower and upper tie plates, grid spacers, and water rods. As described in NEDC-33237P, the GEXL14 correlation was originally developed using full-scale test data obtained from the ATLAS critical power test facility. This facility used an electrically heated mockup of a BWR fuel bundle containing prototypical spacers and operating at conventional BWR flow rates, pressures, and temperatures. The staff approved the use of the GEXL14 correlation for conventional GE14 fuel in the final safety evaluation report (FSER) issued August 3, 2007, for the topical report NEDC-32851P, Revision 2, “GEXL14 Correlation for GE14 Fuel.” Similar critical power tests have been conducted at the Stern Laboratories test facility in Hamilton, Ontario, using a full-scale mockup of a GE14E bundle, with operating conditions expected for the ESBWR. NEDC-33413P, “Full Scale Critical Power Testing of GE14E and Validation of GEXL14,” issued March 2008, documents these tests and their statistical evaluation.

The critical power and pressure drop tests conducted for a simulated GE14E fuel bundle validate the use of the GEXL14 correlation and demonstrate the adequacy of the established GEXL14 statistics for the GE14E fuel. NEDC-33413P provides the details of the test facility, test matrix, test results, and GEXL14 statistical analysis.

To evaluate the effect of design differences between the GE14E and GE14 fuel, the applicant has used the steady-state sub-channel analysis computer code COBRAG. COBRAG is used to predict bundle critical powers and dryout locations, bundle averaged and planar local void fractions, and bundle pressure drops in BWR fuel bundles. GEH submitted the COBRAG code to the staff to enable the staff to assess the sensitivity of the GE14E fuel design to spacer locations and part-length rod height within the fuel bundle. The staff validated the applicability of the code independently by comparing the code’s predictions to benchmark fuel data. The validation results confirmed the GEH claim that the COBRAG code is an appropriate computational tool as applied to the adjustment of the GEXL14 correlation additive constants, which are used to account for variation in power between fuel rods.

Topical report NEDC-33237P provides a detailed description of the studies performed for the assessment of differences in total heated length of the fuel assemblies, grid spacer separation, and part-length rod height and presents a statistical determination of the critical power correlation uncertainties. The overall correlation uncertainty, which includes both measurement and calculation uncertainties, will be applied to all ESBWR applications where the correlation is used.

In DCD Tier 2, Revision 9, Section 4.4, the applicant described how the ESBWR meets GDC 10 and GDC 12 and other acceptance criteria of SRP Section 4.4 by direct reference to the fuel design acceptance criteria provided in DCD Tier 2, Revision 9, Appendix 4B, and NEDC-33237P. These references provide thermal-hydraulic parameters and limits related to neutronic and thermal-hydraulic aspects of the fuel design. Section 4.2 of this report documents the staff evaluation of DCD Tier 2, Revision 9, Appendix 4B.

A brief summary of technical information is provided below by subject.

4.4.2.1 *Critical Power*

The thermal-hydraulic design of the core establishes the thermal-hydraulic safety limits for use in evaluating the safety margin in accordance with GDC 10. The margin to SAFDL is maintained during normal operation when the OLMCPR is greater than the safety limit minimum critical power ratio (SLMCPR) and the LHGR is maintained below the MLHGR limit(s). The MCPR is the minimum CPR of all of the fuel bundles. The CPR for any bundle is the ratio of the bundle power that would result in transition boiling to the current bundle power. Therefore, the bundle with the smallest CPR has the smallest margin to transition boiling. The CPR is a function of several parameters; the most important are bundle power, bundle flow, the local power distribution, and the details of the bundle mechanical design.

Section 5.13 of topical report NEDC-33237P discusses this in further detail. The limits are determined by analysis of the most severe AOOs and, considering uncertainties, provide reasonable assurance that no significant fuel damage results. Thermal margin is stated as the minimum value of the CPR that corresponds to the limiting fuel assembly in the core. The design requirement is based on a statistical analysis that demonstrates that, for AOOs, at least 99.9 percent of the fuel rods would be expected to avoid reaching boiling transition.

NEDC-33237P presents the results for the bundle critical power performance. Full-scale GE14 fuel test data are used to support the development of a critical power correlation for the ESBWR fuel, GE14E. An analytical approach is provided to adjust the GE14 test data to account for the GE14E design differences, including the overall heated length of the fuel assembly, the part-length rod height differences, and the grid spacer separation differences. This approach is demonstrated to be conservative by the confirmatory critical power tests conducted on a simulated GE14E fuel bundle.

4.4.2.2 *Fuel Cladding Integrity Safety Limit*

The fuel cladding integrity safety limit (FCISL) is specified such that no significant fuel damage is calculated to occur during normal operation and AOOs. Although it is recognized that the onset of boiling transition would not result in damage to BWR fuel rods, a calculated fraction of rods expected to avoid boiling transition has been adopted as a safety limit. The FCISL is defined as the fraction (percent) of total fuel rods that are expected to avoid boiling transition during normal operation and AOOs. A value of 99.9 percent provides assurance that SAFDLs are met. NEDC-33237P provides the FCISL results, along with an evaluation of the uncertainties applicable to the ESBWR design. The statistical analysis model used produces a CPR map of the core, based on steady-state uncertainties that are coupled to the TRACG ΔCPR/initial critical power ratio (ICPR) results to develop the OLMCPR. Section 5.13 of NEDC-33237P gives details of the calculation procedure.

4.4.2.3 *Operating Limit Minimum Critical Power Ratio*

A plant-unique OLMCPR is established to provide adequate assurance that the FCISL for that plant is not exceeded during normal operation and any AOO. By operating with the MCPR at or above the OLMCPR, the FCISL for that plant is not exceeded during normal operation and AOOs. Section 5.13 of NEDC-33237P presents a detailed discussion of the OLMCPR calculation methodology.

In addition, NEDC-33237P evaluates the fuel bundle critical power performance. This report uses full-scale test data for a conventional BWR GE14 fuel bundle to justify that the same critical heat flux (CHF) correlation can be applied conservatively to the GE14E ESBWR fuel.

LTR NEDC-33413P describes the tests conducted on a simulated GE14E fuel bundle to confirm that the GEXL14 correlation, with adjustments to the additive constants to account for differences between GE14 and GE14E fuel, can be used for CPR determination of the ESBWR fuel. The tests demonstrate that the correlation is conservative when applied to GE14E fuel over the expected ESBWR operating range.

4.4.2.4 *Void Fraction*

The void fraction in a BWR fuel bundle has a strong effect on the neutron flux and power distribution. The ESBWR design calculations use an empirical correlation based on the characteristic dimensions of the fuel bundle and the hydraulic properties of the flow in the bundle. The 3D core simulator code (PANAC) and the steady-state thermal-hydraulic calculations utilize the GEH void correlation. Section 21.6 of this report discusses the staff evaluation of the TRACG program for ESBWR transients. LTR NEDC-33239P discusses the three-dimensional, quasi-steady-state core simulator model, PANAC11. The staff evaluation of the core simulator code is presented in the SER for GEH LTRs NEDC-33239P and NEDE-33197P.

The Findlay-Dix correlation (Proprietary Report NEDE-21565, "New BWR Void Fraction Correlation," issued January 1977) is used in the three-dimensional core simulator and in steady-state thermal-hydraulic calculations. This approach is also described in NEDC-32084P-A, Revision 2, "TASC-03A, A Computer Program for Transient Analysis of a Single Channel."

The TRACG computer program, used for transient analyses of LOCAs, ATWS, and AOOs, employs a drift flux, interfacial shear model, which is described in NEDE-32176P, Revision 3, "TRACG Model Description," issued April 2006. NEDE-32177P, Revision 2, "TRACG Qualification," issued January 2000, discusses the qualification of the TRACG program. The SER for NEDE-33083P, Supplement 3, describes the staff's review and approval as it relates to ESBWR transient analysis.

4.4.2.5 *Core Pressure Drop and Hydraulic Loads*

The TRACG program has been used to calculate the reactor internal pressure drop and hydraulic loads during normal operation and all AOOs, infrequent events, and accidents (e.g., LOCAs). The total pressure drop consists of friction, local, elevation, and acceleration terms. The TRACG model of the reactor vessel internals consists of radial and axial nodes that represent the boundaries of internal components. They are connected by flow paths with appropriate resistance and inertial characteristics. TRACG solves the equations of conservation of mass and energy for each node, along with the momentum equation, to give depressurization rates and local pressures. Internal component loads are then calculated from the pressure differences. Approved LTR NEDC-33083P-A, Revision 0, "TRACG Application for ESBWR," issued March 2005, discusses the TRACG program flow and pressure drop models for the ESBWR design in detail. NEDE-32176P provides the theoretical development and model description. NEDE-32177P describes the TRACG program qualification.

The friction pressure drop component is calculated as a conventional two-phase pressure drop, with a single-phase friction factor and a two-phase friction multiplier. Full-scale rod bundle pressure drop data from LTR NEDC-33238P, Revision 0, "GE14 Pressure Drop Characteristics," were used to validate these friction factors for GE14 fuel components, including upper and lower tie plates, grid spacers, water rods, and part-length rods. The local pressure drop component is defined as the irreversible pressure loss associated with an area change, such as an orifice, lower tie plate, and grid spacers. It is calculated in a manner similar to the friction pressure drop, except that the local loss coefficient, K, replaces the friction coefficient. The coefficients are determined by tests, as documented in NEDC-33238P.

Additional pressure drop tests were performed using a mockup of the GE14E fuel bundle, with operating conditions expected for the ESBWR. These tests were documented in LTR NEDC-33456P, "Full-Scale Pressure Drop Testing for a Simulated GE14E Fuel Bundle," issued March 2009. The ESBWR fuel bundle-specific (GE14E) critical power and pressure drop testing was performed to better characterize the thermal-hydraulic performance of the GE14E fuel, which is shorter than the conventional GE14 fuel used in currently operating BWRs. The differences include the active fuel length, the number and axial location of the fuel rod spacers, and the axial length of the part-length rods. Also, there are differences in nominal operating conditions. The nominal bundle power and flow for the ESBWR are lower than those for the current operating fleet BWRs. The spacer loss coefficients are determined for the GE14E fuel in the ESBWR application from the pressure drop test data.

The elevation pressure drop component is determined by a conventional approach, accounting for the density change over a given height. The equation appears in DCD Tier 2, Revision 9, Section 4.4.2.3.3. The density term is the average mixture density, with liquid and vapor components of the two-phase fluid, weighted by the void fraction, which is determined by the drift-flux model incorporated in TRACG.

The acceleration pressure drop component is a reversible pressure change that occurs when an area change is encountered, and it is an irreversible loss when the fluid is accelerated through the boiling process. DCD Tier 2, Revision 9, Section 4.4.2.3.4, presents the equations used in TRACG.

NEDC-33083P-A discusses detailed core pressure drop methodology for the ESBWR. DCD Tier 2, Revision 9, Table 4.4-1a and 4.4-1b provide thermal-hydraulic design characteristics of the ESBWR reactor core and compare these to typical BWR/6 and ABWR values.

4.4.2.6 *Core Coolant Flow Distribution*

Based on the prediction of core pressure drop, the distribution of flow into the fuel channels and the core bypass regions is calculated using the TRACG program. The core coolant flow distribution forms the basis of the prediction of steady-state and transient critical power and void fraction. TRACG treats all fuel channels as one-dimensional (axial), while the vessel is modeled as a three-dimensional component. The bundle pressure drop evaluation includes frictional, local, elevation, and acceleration losses. The pressure drop methodology has been qualified to test data in NEDE-32177P. NEDC-33083P-A discusses the TRACG program flow and pressure drop models for the ESBWR design in detail. TMI Action Item II.F.2 of NUREG–0737 requires instrumentation, such as level sensors, for the indication of inadequate core cooling. DCD Tier 2, Revision 9, Table 1A-1, discusses the proposed RPV level instrumentation. DCD Tier 2, Revision 9, Chapter 7, discusses the instrumentation and control systems in detail. The ESBWR design provides for the detection of conditions indicative of inadequate core cooling by a direct water-level instrumentation system. Both wide-range and fuel zone instruments measure the coolant level in the RPV. The RPV water level is the primary variable indicating the availability of adequate core cooling. Adequate redundancy is provided from the bottom of the core support plate to the centerline of the main steamlines.

The ESBWR is designed in accordance with the most recent Revision 4 of RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

4.4.2.7 *Fuel Heat Transfer*

The heat transfer model must accurately predict heat transfer between the coolant, fuel rod surface, cladding, gap, and fuel pellet in the evaluation of core and fuel safety criteria. Conventional methods and assumptions are employed in the TRACG program, which is used for transient analyses of LOCAs, ATWS, and AOOs. NEDE-32176P, NEDE-32177P, and NEDC-33083P-A discuss the TRACG heat transfer models in detail.

The TRACG program includes standard heat transfer regimes (single-phase liquid or vapor), nucleate boiling, CHF, transition boiling, film boiling, and condensation with and without the effect of non-condensables. The program provides correlations for transition between different heat transfer regimes. The correlations for different regimes are standard, well-accepted correlations from the literature. However, for CHF, TRACG uses the proprietary GEXL correlation (GEXL), based on the critical quality concept for normal flows. The NRC has approved the GEXL correlation for specific fuel designs, including GE14, which provides the basis for the GE14E design in NEDC-33240P.

During normal operation and AOOs, convection and nucleate boiling are the most significant heat transfer mechanisms between the coolant and fuel rod surfaces. The applicant used the Dittus-Boelter correlation for the single-phase convective heat transfer for both fuel design (in the core simulator code, PANAC) and systems analyses (in the TRACG code). For nucleate boiling, the applicant used the Jens-Lottes correlation for fuel design (in the core simulator code) and the Chen correlation for systems analyses (performed using TRACG). These three correlations are widely accepted in the nuclear industry for rod bundle heat transfer. The fuel rod thermal-mechanical design analysis program, GSTRM, incorporates the same heat transfer models, as discussed in DCD Tier 2, Revision 9, Section 4.2. NEDC-33239P discusses the core simulator code, PANAC.

4.4.2.8 *Maximum Linear Heat Generation Rate*

The adequacy of MLHGR limits is evaluated for the most severe AOOs to provide reasonable assurance that no fuel damage results during AOOs. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs. Additional discussion appears in the Section 4.3 of this report. DCD Tier 2, Revision 9, Section 15.2, provides the AOO results.

4.4.2.9 *Core Power Operating Map*

DCD Tier 2, Revision 9, Section 4.4.4.3, states that the core power-flow map is a single line, and there is no active control of the core flow at a given power level. The applicant provided a core power-feedwater temperature operating map in DCD Tier 2, Revision 9, Figure 4.4-1, which increases operating flexibility. DCD Tier 2, Revision 9, Sections 10.4.7.2.2.3 and 7.7.3, respectively, discuss the system hardware and control system required to develop and implement such an operating domain. LTR NEDO-33338, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," issued May 2009, presents a detailed discussion and analysis. Section 15.1 of this report discusses the staff review of NEDO-33338.

4.4.2.10 *Inadequate Core Cooling Monitoring System*

The ESBWR inadequate core cooling (ICC) monitoring system is discussed in DCD Tier 2, Revision 9, Appendix 1A, "Response to TMI Related Matters." TMI Item II.F.2 in Table 1A-1 (TMI Action Plan Items) addresses this issue as it relates to the ESBWR. The ESBWR ICC monitoring system provides direct water-level instrumentation, using both wide-range and fuel zone instruments. The four divisions of wide-range instruments cover the range from above the core to the main steam lines. The four channels of fuel zone instruments cover the range from below the core to the top of the steam separator.

4.4.2.11 *Loose Parts Monitoring System*

The applicant has withdrawn the LPMS from the ESBWR design certification for the reasons given in response to RAI 4.4-7, RAI 4.4-8 and RAI 4.4-9. Section 4.4.3 of this report discusses the staff evaluation. Based on the applicant's responses, RAI 4.4-7, RAI 4.4-8, and RAI 4.4-9 are resolved.

4.4.2.12 *Testing and Verification*

Chapter 14 discusses the testing and verification techniques to be used to ensure that the planned thermal and hydraulic design characteristics of the core have been provided and will remain within required limits throughout the core lifetime.

4.4.3 Staff Evaluation

The following presents the staff evaluation of core thermal-hydraulic topics discussed in DCD Tier 2, Revision 9, Section 4.4.

4.4.3.1 *Critical Power*

DCD Tier 2, Revision 9, Section 4.4, references LTR NEDC-33237P to justify the use of the GEXL14 correlation for ESBWR fuel (GE14E) applications and to describe the determination of the overall correlation uncertainty. This document describes the application of the GEXL14 critical power correlation to ESBWR fuel (GE14E) and the supporting analyses performed to quantify and subsequently account for the effect on critical power of the differences between GE14 for the conventional BWRs and GE14E for the ESBWR. The GEXL14 critical power correlation for conventional GE14 10x10 fuels was developed using data obtained from the ATLAS critical power test facility. GE14 fuel is currently being used in operating BWRs. A significant and successful operating experience base has been developed for BWRs using GE14 fuel operating at the original design rated power and those operating at extended power uprate. Because of the similarity between the conventional BWR and ESBWR versions of GE14, the applicant proposed to use the GEXL14 correlation for ESBWR applications, with adjustment for the geometry differences between the two versions of GE14.

First, the ATLAS critical power data for the conventional BWR version of GE14 is adjusted because of the shortening of the heated length of the fuel assembly. A COBRAG subchannel computer program analysis model of GE14, previously qualified based on the ATLAS GE14 critical power data, is then used to quantify the effect of the geometry differences between the two GE14 versions on the critical power performance of the ESBWR version of GE14.

The staff's review of NEDC-33237P, which includes an assessment of the critical power evaluation method described by the applicant in DCD Tier 2, Revision 9, Section 4.4, appears in detail in the safety evaluations for NEDC-33237P and NEDC-33413P. NEDC-33237P, Revision 4, incorporates RAI responses based on previous revisions and references test report NEDC-33413P to confirm the applicability of the GEXL14 correlation to GE14E fuel.

The staff performed confirmatory analysis of the COBRAG code studies used by the applicant to adjust the GEXL14 correlation predictions to account for the differences in grid spacer separation and part-length rod height between the GE14E fuel used for the ESBWR and the GE14 fuel used in operating BWRs. In RAI 4.4-25, the staff requested the applicant to provide the COBRAG program and input decks so that sensitivity studies could be performed with the closer grid spacer separation and the shorter part-length rod height of the ESBWR. The studies confirm the applicant's statement that the effects on critical power of grid spacer separation and shorter part-length rod height offset each other. Based on the applicant's response and the confirmation of spacer separation and shorter part length rods relation, RAI 4.4-25 is resolved.

In RAI 4.4-1, the staff requested that GEH provide detailed information regarding the following:

- Analyses and testing performed to demonstrate compliance of the ESBWR with regulations

- The means by which the design addresses the regulatory guidance outlined in SRP Section 4.4

- Justification of the applicability of traditional computational methods (if used) to the ESBWR

- Justification of the applicability of new computational methods (if used) to the ESBWR, as well as differences between new methods and traditional methods

In part (a) of RAI 4.4-1, the staff requested the applicant to state if any analyses or tests are necessary to demonstrate compliance with the regulations. Part (a) also asked that GEH discuss the theoretical or experimental basis, the method used, the assumptions and boundary conditions, the limitations, and the results as applied to the ESBWR design. The staff agreed that the critical power performance of the GE14E fuel will be similar to that of the GE14 fuel already used in operating BWRs, since the bundle components are similar. However, the staff believed that the thermal-hydraulic response characteristics of the shorter overall length of the GE14E assemblies, and, in particular, the shorter part-length rod height, may contribute to variation in the critical power correlation uncertainties which cannot be accurately determined by computer code (COBRAG) assessment alone. Therefore, the staff requested that the applicant submit a proposed CHF (critical power performance) test matrix for the GE14E fuel and submit proposed ITAAC to ensure that CHF testing is conducted to validate the use of the GEXL14 correlation for ESBWR application before initial core loading.

In response (to part (a) of RAI 4.4-1), the applicant stated that no new testing is necessary to demonstrate compliance of the ESBWR core thermal and hydraulic design with regulations. The applicant further stated that the analysis methods are applicable to the ESBWR, as discussed in the revised LTR NEDC-33237P. The applicant discussed how the ESBWR design meets the regulatory guidance in Section 4.4 of the SRP by complying with GDC 10 and GDC 12. GDC 10 requires that the reactor core be designed such that fuel design limits will not be exceeded. The ESBWR conforms to GDC 10 by establishing a MCPR such that at least 99.9 percent of the fuel rods in the core would avoid boiling transition during normal operation or AOOs. DCD Tier 2 Section 4.4.1.1.1 discusses conformance with GDC 10. GDC 12 requires that when fuel design limits are exceeded, power oscillations either cannot occur or can be reliably and readily detected and suppressed. The applicant stated that the stability evaluation in DCD Tier 2, Appendix 4D sufficiently addresses GDC 12. The ESBWR is designed to maintain stability during normal operation, as well as during AOOs. As a backup, the ESBWR maintains the ability to detect and suppress instability. The ESBWR complies with GDC 12 by implementing design criteria for the decay ratio in the form of a stability map.

Additionally, the applicant stated that the TRACG code is used for the systems analysis of the ESBWR and that a core simulator code is used for the ESBWR core design. It further stated that the responses to several of the RAIs concerning Section 4.4 show that the models in both codes apply. In addition, the NRC has already approved the applicability of TRACG to the ESBWR for LOCA and stability analyses. The staff documented its review of TRACG application for an ESBWR LOCA in the "Addendum to the Safety Evaluation Report for NEDC-33083P-A, 'Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design,'" and the "Addendum to the Safety Evaluation Report by the Office of New Reactors Application of the TRACG Computer Code to Thermal-Hydraulic Stability Analysis for the ESBWR Design NEDE-33083P, Supplement 1.'" The staff addressed the applicability of TRACG to ESBWR ATWS and AOO analyses in the "Safety Evaluation by the Office of New Reactors "TRACG Application for ESBWR Anticipated Transients Without Scram Analyses," NEDE-33083P, Supplement 2, Revision 2" and in the "Safety Evaluation for the TRACG Application for ESBWR Transient Analysis NEDE-33083P, Supplement 3, Revision 1." Also, the applicant stated that no new computation methods are used for the ESBWR.

The staff found the response to RAI 4.4-1 acceptable, with the exception of the GEH position that no new testing was necessary for the GE14E fuel because of the difference in length between the GE14 fuel assemblies (for which testing was performed) and the GE14E fuel assemblies. This exception resulted in RAI 4.4-1 S01. This RAI requested that the applicant submit a proposed CHF test matrix for the GE14E fuel and the corresponding proposed ITAAC to ensure that CHF testing is conducted before initial fuel load to validate the use of the GEXL14 correlation.

The applicant's response to RAI 4.4-1 S01 referred to full-scale GE14E testing that has been performed and provided NEDC-33413P. At the time the response was transmitted, the results of the GE14E testing were still being analyzed. Additionally, the applicant responded that "As testing has been performed sufficient to confirm the adequacy of GEXL14, it is not necessary to construct an ITAAC."

The staff's review of the response to RAI 4.4-1 S01, led to RAI 4.4-1 S02 where the staff requested further justification for the following four items:

- Explain the axial power distributions used for the tests.

- Provide justification for testing only at 6.9 megapascals (MPa) (1,000 pounds-force per square inch absolute [psia]).

- Provide the statistical assessment of the GEXL correlation uncertainty for GE14E fuel.

- Use the GE14E test data to show that the R-factor calculation methodology is applicable to the ESBWR.

The applicant's response, discussed each of the above items.

The staff review of the responses to these RAIs concluded that the applicant has sufficiently addressed RAI 4.4-1 and supplements S01 and S02 to this RAI by the incorporation of revisions to NEDC-33237P and NEDC-33413P. Therefore, based on the applicant's responses, RAI 4.4-1 and its supplements S01 and S02 are resolved.

In RAI 4.4-5, the staff requested that the applicant describe the applicability of the bundle critical power performance method to the ESBWR design. DCD Tier 2, Revision 9, Section 4.4.2.1.1, refers to topical report NEDO-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application," issued January 1977. The staff requested that the conditions and limitations applicable to its use for the ESBWR design be specified. In response, the applicant indicated that NEDC-33237P discusses the applicability of GEXL to GE14E fuel, and a revision to that report will provide additional information. The applicant revised NEDC-33237P to include reference to the GE14E bundle-specific test data and analyses documented in NEDC-33413P. This report confirms that the critical power performance method proposed by the applicant is conservative. Therefore, based on the applicant's response, RAI 4.4-5 is resolved.

In RAI 4.4-19, the staff requested a description of the uncertainties referred to in the DCD and a comparison to conventional (operating) BWR uncertainties.

These include:

- Uncertainty ranges of manufacturing tolerances
- Uncertainties in measurement of core operating parameters
- Calculation uncertainties
- Uncertainty in the calculation of the transient ΔCPR/ICPR
- Statistical uncertainty associated with the critical power correlations

In response, the applicant updated Section 5.0 of NEDC-33237P to include a detailed discussion of the uncertainties that contribute to the overall uncertainty in the GEXL14 correlation when applied to the ESBWR. Most of the uncertainties, including pressure, flow, and temperature measurement, are not unique to the ESBWR. The staff reviewed the information provided by the applicant and determined it to be acceptable In that they are not unique to the ESBWR. Therefore, based on the applicant's response, RAI 4.4-19 is resolved.

RAI 4.4-26 summarizes the applicant's commitments from the closed proprietary meeting during the week of June 19, 2006, which include a revision to topical report NEDC-33237P to provide supporting test data and additional discussion of uncertainties. Specifically, the staff requested the ATLAS test data for GE14 fuel that were used in support of the adjustment of the GEXL14 correlation to account for GE14E fuel design differences, along with a more detailed discussion of the uncertainties unique to the ESBWR design and a statistical analysis using a 95/95 confidence level methodology.

The applicant responded by providing a draft revision to topical report NEDC-33237P, which added Appendices A, B, and C. The response provided ATLAS critical power test data for the GE14 fuel that was used for the COBRAG studies and a statistical evaluation, presented in tabular form and adjusted for the truncated length of the GE14E fuel used for the ESBWR. The applicant subsequently incorporated the draft appendices in Revision 1 of NEDC-33237P. Appendix A describes the COBRAG sub-channel analysis. Appendix B provides the ATLAS data for various pressure, mass flux, and inlet sub-cooling conditions, along with the corresponding adjusted critical power. The same table compares the COBRAG results considering grid spacer separation differences and part-length rod height differences of the GE14E fuel. The GEXL14 correlation prediction for the same test conditions appears in a separate column.

In RAI 4.4-26, the staff requested the applicant to explain the discrepancy in Table A-1 of NEDC-33237P, the GEXL14 10x10 COBRAG/ATLAS critical power category, and the supporting database provided in Table B-1. In addition, the staff requested the applicant to provide information for the individual assemblies missing from Table A-2 and, for Table A-5, to state what database was used to derive the ATLAS cosine standard deviation. The staff noted that the applicant should identify the test runs for Table C-1, and in Section A.2, the applicant should explain which data are used and how they are applied. The applicant provided a revision to NEDC-33237P which incorporated the corrections and the additional information. The staff reviewed the revised topical report and conducted its own internal calculations to confirm the applicant's results. The staff concludes that the applicant's statistical analysis regarding the ESBWR critical power correlation and the associated uncertainties are acceptable because they are comparable with those estimated by the staff. Therefore, based on the applicant's response, RAI 4.4-26 is resolved.

4.4.3.2 *Fuel Cladding Integrity Safety Limit Minimum Critical Power*

In conventional BWRs, Technical Specification (TS) 2.1.1 specifies the FCISL as an SLMCPR. The FCISL depends on the operating mode, the reactor steam dome pressure, and the core flow as a percentage of rated core flow. TS 2.1.1 provide a MCPR for allowed operation. If the condition is not met, the reactor must be shut down. The ESBWR TSs are based on the standard BWR/6 TSs in NUREG–1434, "Standard Technical Specifications General Electric Plants, BWR/6," Revision 3, Volume 1. DCD Tier 2, Revision 9, Chapter 16, and the corresponding bases in Chapter 16B provide reactor core safety limits in TSs 2.1.1 and B2.1.1. The FCISL is designed such as 99.9 percent of the fuel rods expected to avoid boiling transition. This differs from the SLMCPR specification for conventional BWRs. In response to RAI 15.0-16, the applicant provided justification for using the number of rods subject to boiling transition as a safety limit to replace the SLMCPR in the ESBWR TS.

The staff disagreed with the applicant's position. The staff concluded that the SLMCPR numerical value should be kept as a safety limit in the TS as in the BWR standard TS. In RAI 15.0-16 S01, the staff requested that the original TS safety limit be restored. In a subsequent DCD revision, the applicant restored the SLMCPR value. Section 15.1.1 of this report presents the detailed staff evaluation. Based on the applicant's response that restored the original TSs, RAI 15.0-16 is resolved.

DCD Tier 2, Revision 9, Section 4.4.3.1.2, refers to Section 6 of LTR NEDC-33237P for a summary of the basis for the representative OLMCPR used for the ESBWR to protect the FCISL. Section 5 of the topical report describes the basis for the uncertainties specific to the ESBWR used in this evaluation.

The staff evaluated the applicant's methodology for the determination of the OLMCPR and FCISL and performed a confirmatory evaluation of the applicant's COBRAG studies to investigate the effects on critical power due to closer grid spacer separation and shorter part-length rod height of the ESBWR GE14E fuel. The staff finds the applicant's general approach acceptable. The staff's SER for NEDC-33237P includes the staff evaluation of this issue.

4.4.3.3 *Operating Limit Minimum Critical Power Ratio (OLMCPR)*

The staff performed a confirmatory evaluation of the COBRAG grid spacer separation and part-length rod height studies used by the applicant to adjust the approved conventional GEXL14 correlation for the GE14E fuel differences. As noted above, the staff finds the applicant's approach acceptable.

The staff performed confirmatory studies to evaluate the use of the GEXL14 correlation for the GE14E fuel of the ESBWR. To assess the GEH application of the COBRAG code for the determination of the effects of closer spacer separation and shorter part-length rods on the ESBWR GE14E fuel, the staff ran a number of parametric cases that approximated the sensitivity studies performed and documented by the applicant in LTR NEDC-33237P. This topical report outlines a procedure by which the ATLAS critical power data collected for the 12-foot GE14 fuel was adjusted using COBRAG to account for geometry differences of the GE14E fuel. This GE14E fuel assembly model was then used to determine the separate and combined effects on critical power of the GE14E spacer locations and part-length fuel rod length.

To approximate the parametric studies presented by the applicant, the staff constructed COBRAG input decks that varied the relevant parameters, starting with the input deck provided with GE14E. When completed, the modified input decks were executed using the GEH version of COBRAG. The resulting output was compared to the results of Run No. 156 in Enclosure 1 of the RAI 4.4-25 response, and differences between the two datasets were found to be less than 1 percent. These small discrepancies were expected and are products of slight modeling differences between the staff's approach and that of the applicant. Nevertheless, the overall trends as presented in the topical report in general, and for Run No. 156 in particular, are in agreement with the staff's results. Differences between the staff's and the applicant's predictions were deemed negligible, consistent, and due to modeling differences. The staff's confirmatory studies do not constitute a formal staff review of the COBRAG program. COBRAG was submitted on the ESBWR docket by Global Nuclear Fuels letter FLN-2007-023, dated July 5, 2007. In a letter dated March 6, 2008, it was withdrawn, since the GEXL14 correlation additive constants applicable to the ESBWR GE14E fuel can be derived from the critical power test measurements for simulated GE14E fuel.

In RAI 4.4-27, the staff requested a discussion of the applicability of an approved topical report, NEDC-32505P-A, Revision 1, “R-Factor Calculation Method for GE11, GE12, and GE13 Fuel,” used for R-factor determination for conventional BWR GE11, GE12, and GE13 fuel bundles, to the ESBWR GE14E fuel bundle design. The staff agreed that the same methodology can be used to calculate the GE14E bundle R-factor, but the applicant should provide confirmation. The applicant responded that no new features affecting the R-factor methodology were introduced with the GE14E fuel design. In response to RAI 4.4-27 S01, the applicant provided additional qualitative and quantitative discussion. The staff stated in RAI 4.4-27 S02 that the staff would be satisfied on the condition that the R-factor must be reassessed and the methodology satisfactorily confirmed when the new critical power performance data are collected for the ESBWR GE14E fuel assembly ITAAC. GEH submitted NEDC-33413P in response to RAI 4.4-27 S02, which summarizes the test results for GE14E bundle-specific critical power tests and provides confirmation that the R-factor methodology used is acceptable. Additionally, the tests confirm that the method proposed by the applicant to evaluate the critical power performance of GE14E fuel is conservative. Since the GE14E fuel bundle-specific tests have been successfully completed, there is no longer a need to establish fuel critical power performance ITAAC. Therefore, based on the applicant’s response, RAI 4.4-27 is resolved.

The staff issued RAI 4.4-28, to request a discussion relating the range of the ATLAS test conditions to expected ESBWR operating conditions and an explanation of the treatment of the electrically heated rods used in the GE14 bundle ATLAS tests that were previously found to influence the result because of magnetic biasing. This RAI requested a discussion of any correction made for this effect. The applicant responded that no adjustment had been made to account for the magnetic biasing attributed to the electrically heated rods of the ATLAS facility and that it would revise Table 4-2 of NEDC-33237P to include the studies that will account for the potential magnetic bias in the ATLAS GE14 critical power data. The applicant submitted the revision of the topical report with no change. In response to RAI 4.4-28 S01, the applicant explained that a conservative adjustment to the GEXL correlation additive constants has been applied to account for the bias in a manner consistent with that applied to operating BWRs and therefore no revision of table 4-2 is required. The staff accepts this explanation. Therefore, based on the applicant’s response, RAI 4.4-28 is resolved.

The staff issued RAI 4.4-29, to request a discussion of the differences in the tested ranges of: pressure, mass flux, inlet subcooling, and R-factor for a GE14 fuel bundle with regard to the ESBWR operating range. In response, the applicant proposed a revision to topical report NEDC-33237P, which includes a table indicating that the range of GEXL14 applicability bounds and the corresponding (expected) GE14E conditions for the ESBWR. The staff finds the response to be acceptable, and RAI 4.4-29 is resolved.

The staff issued RAI 4.4-30, to request an explanation of the conservatism of the average experimental CPR, using the adjusted correlation, to the measured critical power from the ATLAS tests. The applicant responded that the uncertainty in the correlation is accounted for by application of an overall correlation uncertainty to ensure conservatism. The applicant added a clarification for Table 4-2 regarding this conservatism. The staff finds the response acceptable. Therefore, based on the applicant’s response, RAI 4.4-30 is resolved.

Section 5.14 of NEDC-33237P provides the methodology for determination of the SLMCPR value. The value specified in the TS includes a conservative multiplier to account for the overall GEXL correlation uncertainty. The staff compared the methodology being applied to the ESBWR for consistency with the current operating BWR methodology. The staff finds that the use of the GEXL14 correlation for the determination of the OLMCPR is acceptable.

In RAI 4.4-61, the staff asked the applicant if the GEXL correlation was used to calculate the MCPR in DCD Tier 2, Revision 3, Figures 6.3-7, 6.3-15, 6.3-23, 6.3-31, 4D-22, and 4D-23, which are related to LOCA analyses and startup stability displayed for these events. If so, the applicant should justify the use of this correlation, since it is being used outside its range of applicability. If not, the applicant should describe the correlation being used and its applicability range.

The applicant responded that the TRACG channel component uses a combination of the GEXL correlation for dryout in annular flow, the Biasi correlation for departure from nucleate boiling, and the Modified Zuber pool boiling CHF correlation for low-flow conditions to determine the transition between nucleate boiling and film boiling. The range of applicability of these correlations is discussed in Section 6.6.6.3 of NEDE-32176P. The Modified Zuber correlation is applied below the lower mass-flux limit (100 kilograms per square meter second [kg/m^2-s] which is equivalent to 20.48 pound [mass] per square foot second [lbm/ft^2-sec]) of the Biasi correlation for bubbly or churn flow, which occurs before the transition to annular flow. The TRACG algorithm for determining the “critical power” or “thermal margin” is such that the GEXL correlation is picked for annular flow at higher mass-fluxes (within its range of applicability) and the Modified Zuber or Biasi correlation or their interpolation is picked at lower mass-fluxes (i.e., those outside the range of the GEXL database, but within the range of their applicability). The specific cases of LOCA and startup analyses mentioned in this RAI are discussed below.

LOCA analyses Figures 6.3-7, 6.3-15, 6.3-23, and 6.3-31 of DCD Tier 2, Revision 3, correspond to the feedwater line break, main steamline break, bottom drain line break, and the gravity-driven cooling system (GDCS) injection line break, respectively. Other related figures concerning the static head inside chimney (Figures 6.3-8a, 6.3-16a, 6.3-24a, and 6.3-32a) and peak cladding temperature (Figures 6.3-14a, 6.3-22a, 6.3-30a, and 6.3-38a) show that for all LOCA cases, the ESBWR core is always covered with water and the cladding never heats up. This is consistent with the MCPR values being significantly greater than unity for all LOCA cases, as shown in Figures 6.3-7, 6.3-15, 6.3-23, and 6.3-3 1.

For startup stability analyses, three different heatup rates (50 megawatts [MW], 85 MW, and 125 MW) were used, corresponding to Figures 4D-23 and 4D-24. Initially, there were no voids in the channels, even in the hot bundle. As the heatup proceeds, the RPV pressure (Figure 4D-12) increases, voids start to appear at the exit of the hot bundle (Figures 4D-18, 4D-19, and 4D-20), and the hot bundle exit flow rate (Figure 4D-21) starts to increase. The MCPR is not evaluated until voids appear in the channel or fuel bundle. When the MCPR evaluation starts, TRACG effectively uses the Modified Zuber or Biasi correlation or their interpolation for low mass-fluxes. As the flow rate and pressure increase, TRACG uses the GEXL correlation, as intended, within its range of applicability.

The staff reviewed the submitted information and finds it responsive to RAI 4.4-61; therefore based on the applicant’s response, RAI 4.4-61 is resolved.

The staff presents its detailed evaluation of ESBWR stability in Appendix 4A of this report.

DCD Tier 2, Revision 9, Section 15.2.6, lists the potentially limiting events that establish the OLMCPR as follows:

- Loss of feedwater heating (LOFWH) with SCRRI actuation failure
- Slow closure of one temperature control valve

- Generator trip from the 100-percent rated power conditions assuming only 50 percent of the total turbine bypass system capacity

- Inadvertent startup of all loops of the isolation condenser system

The staff issued RAI 4.4-62, to request that the applicant should revise topical report NEDC-33237P, Revision 2, Sections 5.12 and 6.0, to reflect the text in DCD Revision 3 regarding the limiting event. In subsequent revisions of DCD Tier 2, Chapter 15, and NEDC-33237P, the limiting events are consistent. Then staff also issued RAI 4.4-62 S01 to suggest that the same inconsistency for the Loss of Feedwater heating with SCRRI failure and the Inadvertent Isolation Condenser Initiation (IICI) and to ensure consistent wording between the DCD and NEDC-33237P. The limiting events were altered as a result of design changes, which include addition of the feedwater temperature operating domain, and changes in the component flow loss coefficient resulting from core support plate and other dimensional changes. The OLMCPR for each fuel cycle will be established for the limiting event and documented in the Core Operating Limits Report (COLR) in accordance with the TSs. Based on the appropriate update to NEDC-33237P by the applicant, the staff finds the applicant’s response acceptable, therefore, RAI 4.4-62 and RAI 4.4-62 S01 are resolved.

NEDO-33337, “ESBWR Initial Core Transient Analysis,” issued October 2007, provides the analysis of these events for the initial core. The staff discusses its evaluation in Section 15 of this report. Reanalysis of these events will be performed for reload core designs. The results shall be reported in the COLR, as specified in Section 5.6.3(a) (2) of the ESBWR TS (DCD Tier 2, Revision 9, Chapter 16).

4.4.3.4 *Void Fraction*

The ESBWR is expected to operate at up to approximately 90 percent void for normal steady state operation and AOOs. This is significantly higher than for conventional BWRs. To assess the adequacy of the void fraction correlation used by the applicant, the staff issued RAI 4.4-2, and requested the database used to develop the correlation. In response, the applicant referred to the staff approved licensing topical report NEDE-21565. The response also provided validation data for the expected range of ESBWR operating conditions for simple test geometries and various fuel geometries, including 4x4, 6x6, 7x7, and 8x8. The response to RAI 4.4-2 provided the staff the means to review the relationship between the nodal void and nodal quality and made the following observations regarding the void-quality correlation topical report:

- The report, prepared in 1977, includes data for 4x4, 6x6, 7x7, and 8x8 fuel bundle designs. It does not include test data for newer fuel designs with greater than 8x8 bundles (such as the 10x10 arrangement of the ESBWR fuel). The uncertainty in the correlation related to geometry effects should be addressed for the newer fuel designs.

- Most of the test data were concentrated at approximately normal BWR operating pressure (6.9 MPa [1,000 psia]). A few measurements were taken at lower or higher pressures (2.76 MPa to 9.65 MPa [400 to 1,400 psia]) for the various bundle geometries. The void fraction correlation is based on an extensive database for the expected normal operating pressures and flow rates. Outside the normal range, there is a significant uncertainty associated with extrapolation of the correlation to high or low void fractions.

- The correlation is biased downward by a factor of two weighting with the CISE (Ciencias de la Seguridas test facility) 4x4 fuel bundle data, since the quick-closing valve arrangement of the CISE tests was considered most reliable. Expected differences in results for 10x10 bundles should be addressed.

- No data are available for the counter-current flow regime. Test data should be acquired for this regime, or justification should be provided for not considering this flow regime.

- No transient testing was performed. Transient data should be acquired to confirm the void fraction correlation accuracy in transient conditions, or justification should be provided for not considering transient conditions.

- Some of the test data were skewed by radial peaking of the power distribution. Additional full-scale void fraction data with skewed radial peaking should be acquired, or justification should be provided for not considering this effect.

The staff found several areas of uncertainty in applying the Findlay-Dix void quality correlation to new GEH fuel designs at high void fraction. The staff also evaluated the formulation of the correlation. The correlation is based on a two-fluid semi-empirical model. The staff was not certain of the appropriateness of the correlation for predicting void fraction above its originally qualified range for new designs. Specifically, the staff was uncertain about the ability of the model to adequately account for the effects of entrained liquid droplets in the vapor core for high void fractions, where the liquid droplets represent an increasingly large fraction of the liquid flow. In response to RAI 4.4-2, the applicant referenced the approved TASC-03A code in topical report NEDC-32084P-A, Revision 2, and the approved GESTAR topical report described in a letter dated January 16, 1986, “Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A.” Section 3.1 of NEDE-32177P, Revision 2, discusses the TRACG code qualification of the void fraction determination approach.

In RAI 4.4-2 S02, the staff indicated that the indirect justification provided in the response to RAI 4.4-2 S01 for the void correlation at high void fractions using operating fleet GE14 pressure drop data is not a substitute for actual void fraction measurements. Additionally, the staff proposed to apply the same adder to the OLMCPR imposed on the GE14 fuel (see Letter dated November 3, 2006, “Commitment to Update GE’s Void Fraction Data” and Supplement 1 to NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains”) as a penalty to account for the uncertainty in the void fraction correlation resulting from the lack of high void fraction data until the data are acquired and assessed. The staff requested the applicant to revise NEDC-33237P to document the proposed OLMCPR penalty. The approved version of NEDC-33237P includes a discussion of the imposed thermal margin adder. The SER for NEDC-33237P includes the OLMCPR adder as a condition for approval of the calculation methodology, which must be satisfied for licensing applications. The staff considers the OLMCPR adder a satisfactory response, and therefore, RAI 4.4-2 and its supplements are resolved.

The applicant provided additional qualification data to demonstrate that the range of expected operating void fractions (from 0 percent to 92 percent) for the ESBWR is within the qualification basis of the void fraction methods. To account for the uncertainties in void fraction prediction for 10x10 fuel bundles, the staff considers it necessary to apply an OLMCPR adder as a penalty.

Because the void reactivity coefficient is a strong function of the void fraction (increasing in magnitude with increasing void fraction), and given the specific concerns regarding the void quality correlation listed above and concerns about the efficacy of the core simulator code (PANACEA) in producing reliable nuclear data for use in downstream transient analysis codes where void fractions may exceed 90 percent locally, the staff approval of the PANAC11 methodology for the ESBWR is contingent on an additional margin to the ΔCPR in the OLMCPR determination. An adder of 0.01 to the OLMCPR is consistent with an approximately 0.5-percent additional uncertainty in nodal transient power. The staff requires that an adder of 0.01 be in place for the ESBWR OLMCPR methodology until the capabilities of the Findlay-Dix correlation are demonstrated for modern fuel designs over the range of void expected for steady-state operation and AOOs characteristic of the ESBWR. Additional detailed discussion of the safety evaluation is presented in NEDC-33239P-A Revision 5, and NEDE-33197P-A Revision 3.

4.4.3.5 *Core Pressure Drop and Hydraulic Loads*

To evaluate the method, assumptions, and results used by the applicant to calculate core pressure drop and component hydraulic loads, the staff requested, in RAI 4.4-20, a discussion of the calculation of the reactor internal pressure drop and associated loads for normal and transient operation. The applicant responded that the TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs, infrequent events, and accidents (e.g., LOCAs).

The discussion in DCD Tier 2, Revision 9, Section 3.9.5, pertains to reactor internal components. The fuel assembly, including the fuel rods, is not considered a reactor internal; however, the pressure differences determined in the section are also used to evaluate the hydraulic loads on the fuel assembly. DCD Tier 2, Revision 9, Section 4.2.3, discusses the hydraulic loads and the resulting stresses for the fuel channel. Details of this analysis appear in Section 3.4.1.8 of LTR NEDC-33240P.

The reactor internal pressure differences are calculated by appropriate selection in the TRACG model of axial and radial nodes, connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. Component hydraulic loads are calculated using the transient reactor internal pressure difference and the projected area of the component.

Section 21.6 of this report presents the staff's assessment of the TRACG program for the ESBWR design. The staff has previously approved the LTR on TRACG application for the ESBWR, NEDC-33083P-A, Revision 0, for ESBWR LOCA application. The staff has also approved TRACG for the ESBWR stability analysis, and approval of TRACG for the ESBWR AOO application is part of the ESBWR design certification application. The staff review of NEDC-33083P-A, Revision 0, also includes the TRACG model description topical report, NEDE-32176P, by reference. The staff finds the above discussion acceptable and responsive to the request; therefore, RAI 4.4-20 is resolved.

In RAI 4.4-24, the staff requested further discussion of the pressure drop qualification test data used to develop pressure loss coefficients. In response, the applicant stated that the GE14E fuel design uses hardware identical to that currently used in GE14 fuel assemblies. Therefore, the component local pressure drop characteristics will be the same. Topical report NEDC-33238P provides test results for the GE14 components at various flow rates and power levels. The range of test conditions includes the expected ESBWR operating range. The staff finds this acceptable.

As discussed in Section 21.6 of this report, core pressure drop testing was ranked high on the phenomena identification and ranking table (PIRT) for ESBWR AOOs. The staff finds the application of TRACG for the determination of core pressure drop and hydraulic loads acceptable, provided that the confirmatory items identified in Section 21.6 are satisfied; therefore, based on the applicant's response, RAI 4.4-24 is resolved.

4.4.3.6 *Core Coolant Flow Distribution*

The staff issued RAI 4.4-23, to request a quantitative comparison of pressure drops and flow distributions in the fuel channels and core bypass regions of the ESBWR to those of conventional BWRs, as well as a discussion of the impact of a flow reduction on the MCPR limit. In response, the applicant provided a table comparing core pressure drop, bypass flow, and fuel channel flow characteristics of the ESBWR to a BWR/6 and a BWR/4 plant of conventional design, with recirculation pumps. The ESBWR core diameter is similar in size to that of a BWR/6.

In DCD Tier 2, Revision 9, Tables 4.4-1a and 4.4-1b provide typical thermal-hydraulic design characteristics of the reactor core. The ESBWR design parameters are compared to those of the conventional BWR/6 and the ABWR. The data reflect the differences expected because of the natural circulation design of the ESBWR. The staff finds the comparison responsive to the request and acceptable; therefore, RAI 4.4-23 is resolved.

In addition, the staff issued RAI 4.4-23 S01, to request a calculation of MCPR as a function of percent of flow blockage. The applicant provided the results of the flow blockage calculation in letter dated October 18, 2006. The calculation shows that a significant portion of the initial flow area for the inlet orifice or for the initial flow area of the lower tie plate must be blocked before boiling transition (CPR = 1.0), is reached. The ESBWR lower plenum velocities are lower than those in forced circulation BWRs, which should reduce the chance that foreign material is swept up to the inlet orifice or lower tie plate. The lower velocity also minimizes impingement of debris on the bundle fine screen filters. The staff finds the response acceptable in view of the significant blockage required to cause the core to reach boiling transition, therefore, RAI 4.4-23 S01 is resolved.

The staff issued RAI 4.4-23 S02, to request the list of assumptions made in the calculations presented in the response to RAI 4.4-23 S01. The applicant provided the list of assumptions made in the calculations. The staff reviewed the list and finds the assumptions to be reasonable and the calculation approach conservative. The response provided justification for the limited debris quantity, types, and sizes that can be expected during an ESBWR LOCA. The primary debris pathway from the containment to the reactor internals, the opening above the GDCS pool, is protected by a perforated steel plate. Also, the natural circulation coolant flow velocity will be much lower than that of conventional forced circulation BWR, so the amount of debris entrainment is significantly less than expected for conventional BWRs. The staff finds the response acceptable and is resolved.

Regarding the debris entrainment issue, the applicant committed to addressing compliance with RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident." In addition, the response to RAI 6.2-173 S01, discusses this topic. The safety evaluation in Section 6.2 of this report, presents the staff's evaluation of the debris transport.

In RAI 4.4-23 S03, the staff requested the applicant to address similarities and differences between the ESBWR fuel and cooling systems and to address the applicability of the BWR Owners Group (BWROG) calculations for downstream effects of LOCA-generated debris. The Owners Group calculations demonstrate that acceptable fuel centerline temperatures can be maintained. The ESBWR thermal insulation is limited to the stainless steel reflective metallic type, which significantly minimizes the quantity of debris that can be transported to the reactor vessel. Operating BWR containments include significant quantities of fibrous insulation material, which can contribute to debris blockage. The ESBWR design ensures that the reactor vessel water level can be maintained above the top of active fuel for any postulated LOCA. Even if one or more fuel bundle inlet orifices are blocked, the channels remain water filled, with flow coming from the top.

In RAI 4.4-23 S04, the staff requested that GEH perform a calculation similar to that performed for the BWROG. The staff's concern was that, during the early portion of the transient when the decay heat remains high and rapid outflow of liquid inventory occurs as a result of depressurization of the reactor from the pipe rupture, some fuel rods may lose cooling and fuel damage may occur. The applicant performed TRACG calculations that demonstrate that saturated liquid conditions can be maintained at all times during a LOCA. Substantial thermal margin is calculated. Section 6.3 of this report summarizes the staff evaluation of ECCS performance for a spectrum of postulated line breaks, including the effects of debris blockage. Based on the applicant's responses above, RAI 4.4-23 S03 and RAI 4.4-23 S04 are resolved.

4.4.3.7 *Fuel Heat Transfer*

The staff issued RAI 4.4-3, to request a discussion of the heat transfer bases. The applicant responded that standard and well-accepted heat transfer correlations between the coolant and the rod surfaces are used. Topical reports NEDE-32176P and NEDC-33083P-A describe these correlations in detail. Section 21.6 of this report presents the staff's assessment of the TRACG program heat transfer model for the ESBWR design that has been reviewed and approved by the staff therefore, based on the applicant's response, RAI 4.4-3 is resolved subject to the conditions and limitations listed in the LTR NEDC-33083P-A.

4.4.3.8 *Maximum Linear Heat Generation Rate (MLHGR)*

The MLHGR is the maximum local heat generation rate (more specifically, the fuel rod with the highest surface heat flux at any nodal plane in a fuel bundle in the core). The MLHGR operating limit depends on the bundle type, and Section 4.2 of this report evaluates the determination of this limit. ESBWR TS 3.2.1 specifies the LHGR. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel will not occur during any AOOs. The SER for NEDC-33239P provides a detailed evaluation of the LHGR determination as a function of core power distribution, which is dependent on the nuclear flux shape. The staff finds the applicant's method for determination of LHGR limits acceptable.

4.4.3.9 *Core Power/Flow Operating Map*

As stated in RAI 4.4-63, the staff noted that in DCD Tier 2, Revision 3, Section 4.4.4.3, the applicant added a statement that a core power-feedwater temperature operating map is envisioned. Previous revisions of the DCD had stated that the core power-flow map is only a single line, and there is no active control of the core flow at a given power level. In addition, DCD Tier 2, Revision 3, Section 4.4.4.4, "Temperature-Power Operating Map," states, "Not Applicable to the ESBWR." The staff expected that the applicant would revise this section to reflect the change to a temperature-power operating map. In NEDO-33338, the applicant provided additional information on the proposed use of feedwater temperature variations to maneuver reactor power. The applicant also added Figure 4.4-1, "Typical ESBWR Core Power-Feedwater Temperature Operating Domain/Map." Chapter 15 of this report discusses the staff evaluation of NEDO-33338 that has been reviewed and approved by the staff. Subsequent DCD revisions incorporate changes to the text in Section 4.4.4.3, which satisfactorily address the staff's concerns stated in RAI 4.4-63. Therefore, based on the applicant's response and the staff approval of NEDO-33338, RAI 4.4-63 is resolved.

4.4.3.10 *Inadequate Core Cooling Monitoring System*

The staff issued RAI 4.4-21, to request a description of the ESBWR ICC monitoring system to satisfy the requirements of SRP Section 4.4. The applicant responded by providing an additional section to the DCD that refers to DCD Tier 2, Table 1A-1 (TMI Action Plan Item II.F.2). The staff reviewed the revised table and finds it acceptable with respect to the thermal-hydraulics detection capability of the system. Chapter 7 discusses the instrumentation and control room display aspects of the system. Additional staff evaluation of the ICC system appears in Section 20.4 of this report. Based on the applicant's response, the added section in the DCD and Table 1A-1, RAI 4.4-21 is resolved.

4.4.3.11 *Loose Parts Monitoring System (LPMS)*

The staff issued RAI 4.4-7 to RAI 4.4-9 (and corresponding S01s) requesting GEH provide information regarding the implementation and operation of the loose parts monitoring system. In response, the applicant informed the staff that it intended to delete the LPMS from the ESBWR design and provided a basis for doing so. In addition, it stated that "small metallic filings and other similar debris could contribute to fuel cladding damage, but the LPMS would not detect this class of debris, and the industry has installed debris filters into the fuel support pieces which may reduce fuel cladding damage due to fretting." The applicant further noted that the ESBWR design incorporates debris filters and that all fuel supplied by the applicant has a filter (at the bottom) to prevent debris from entering the bundle.

The staff also issued RAI 4.4-7 S02, RAI 4.4-8 S02, and RAI 4.4-9 S02 to request supplemental information to assist the staff in determining whether deletion of the LPMS from the ESBWR design is acceptable. The request included a detailed discussion of: (1) the design of ESBWR debris filter, (2) the maximum size of debris that can pass through the filter, and (3) adverse impacts on cladding and other components in the core caused by the debris that passes through the filter. Additionally, the staff requested an assessment of the adverse impact on ESBWR safety-related systems and components caused by the debris that originates downstream of the filter. The safety assessment was to address the potential for physical damage and flow blockage, particularly focusing on the ESBWR unique features, including the potential for flow blockage of natural circulation and gravity-driven flow lines.

The staff specified that the response was to include, but not be limited to, the following ESBWR components:

- Depressurization valves (DPVs)
- Main steam isolation valves (MSIVs)
- Isolation condenser system-tubes and valves
- SLCS-injection lines
- GDCS-injection lines and valves
- CRD system

The staff also requested a detailed explanation and demonstration that the ESBWR can be safely operated without an LPMS. A systematic analysis of all systems and components in the RPV and the connected systems is required to justify the deletion of the LPMS.

In response to these supplemental RAIs, the applicant stated that the same debris filters used in GE12 fuel are integrated into the lower tie plates of each fuel bundle in the ESBWR. Water must pass through the flow holes before entering the fuel bundles. The LPMS would not detect objects small enough to pass through the filters.

Additionally, the applicant indicated that it is expected that licensees will employ a rigorous foreign materials exclusion program to prevent external sources of loose parts. They will also conduct underwater visual vessel internals inspections during outages to check the structural integrity of reactor components. This will also provide opportunities to find loose parts in the area where inspections are performed. In RAI 4.4-9 S02, the staff requested that the applicant incorporate into the DCD the justification for not providing an LPMS for the ESBWR. In addition, the staff requested an update to DCD Table 1.9-21 for RG 1.133, indicating that the LPMS will be deleted from ESBWR design. The applicant responded with proposed revisions to the DCD as requested in the RAIs. Therefore, based on the applicant's responses, RAI 4.4-7, RAI 4.4-8, and RAI 4.4-9 (and corresponding S01 and S02) are resolved.

The ESBWR design considers important aspects such as material selection and analysis for internal components to prevent failures, and it uses proven design methods to fasten components. In addition, in accordance with RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Pre-Operational and Startup Testing," the applicant plans to install instruments on components during initial startup of the lead ESBWR plant as part of a program to measure the flow-induced vibration of critical components. The type and location of instrumentation is established by detailed evaluations of the RPV components, using prior test data and analyses to determine susceptibility to flow-induced vibration. This minimizes the possibility for internal sources of loose parts in the RPV because of vibration.

The only two systems that provide fluid flow directly into the RPV during normal operation are the feedwater and CRD system. The feedwater system uses temporary strainers as a precautionary measure to ensure that loose parts that may have been left during the construction phase do not enter the RPV. These temporary strainers and any debris collected are removed after the first cycle. Also, the feedwater sparger inside the RPV provides a difficult path for large objects to pass through and enter into the RPV. Objects entering the feedwater spargers must pass through a 5-cm (2-in.) short-radius elbow and then pass through a nozzle with a maximum diameter of (4.8 cm [1.875 in.]) to enter the RPV. Objects that are restricted within the feedwater spargers do not adversely affect the operation of the plant or the feedwater spargers. This minimizes the opportunity for loose parts to enter the RPV through the feedwater system.

For the CRD system, purge water flow enters from the bottom of the FMCRD through a 3.2-cm (1.25-in.) line. Because of the restricted flow paths within the drive, only small objects that an LPMS would not detect could possibly enter the drive. GEH concurs with the Electric Power Research Institute (EPRI) assessment that a loose part is not likely to enter into the CRD and restrict its operation. NEDC-32975P-A, Revision 0, "Regulatory Relaxation for BWR Loose Parts Monitoring Systems," issued February 2001, states the following:

> The EPRI report also stated that loose parts do not, in general, affect CRD operation, because of the torturous path required for loose parts to enter the

CRD guide tube. From the upper plenum, the clearance between the fuel channel and the top of the guide tube is small and movement of any loose parts would be counter to core flow. From the lower plenum, access to the CRD guide tube by metallic parts is effectively prevented by the integrity of the guide tube and the core flow patterns that exist in the fuel bundle and bypass regions. Any debris which enters a CRD guide tube is unlikely to have sufficient mechanical strength to interfere with the operation of the CRD.

In the safety evaluation of the topical report NEDC-32975P-A, the staff also agreed with the EPRI report's evaluation that small loose parts or debris from the lower plenum will probably not impede CRD operation because of the difficult flow path. The staff further stated that small loose parts and debris could enter the CRD during refueling, but the LPMS will not likely detect this class of debris.

The applicant stated that in the event of a loose part entering the vessel, the ESBWR design is capable of performing its safety-related functions. The plant has been designed with multiple DPVs and safety/relief valves (SRVs). In the event a DPV or SRV is restricted, the remaining DPVs and SRVs can accomplish the task of blowdown. The plant has been designed with redundant MSIVs. If one of the series MSIVs becomes restricted, the remaining MSIV can accomplish the task of isolation. The isolation condenser system has four independent trains. If one of the trains is restricted, the remaining three trains can accomplish the task of heat removal. The SLCS has two independent trains. Each train has an injection line that branches into two sets of three injection nozzles within the core shroud. If one of the injection nozzles becomes restricted, the remaining 11 nozzles can accomplish boron injection. The GDCS has four independent trains. If one of the trains is restricted, the remaining three trains can accomplish the task of supplying inventory for a LOCA. Design and testing are performed appropriately to ensure that loose parts are not generated internally. Foreign materials exclusion programs are performed to limit externally generated loose parts from entering the reactor coolant pressure boundary (RCPB). Underwater in-vessel visual inspections are performed to detect cracking of components that can become potential loose parts. In addition, with the redundancy in the design of the safety systems, GEH concluded that the ESBWR is capable of performing its safety-related functions without an LPMS. The staff concurs with the GEH justification described above for not including the LPMS in the ESBWR design.

It may also be noted that the staff agrees with the BWROG regarding the deletion of the LPMS from the currently operating plants. The staff stated that the safety benefits of the LPMS do not appear to be commensurate with the safety benefit and the associated radiation exposure of plant personnel.

4.4.3.12 *Testing and Verification*

The staff reviewed the ITAAC listed in DCD Tier 1, Table 2.1.1-3, pertaining to the RPV and internals. These ITAAC are intended to ensure that the as-built component dimensions and arrangement are consistent with the design analyses. The staff also reviewed DCD Tier 1, Table 2.1.2-3 for impact on core thermal-hydraulic design. Parameters that have been used in the design analyses for natural circulation flow, such as pressure loss coefficients, component free volumes, geometry, hydraulic diameters, and flow areas, will be confirmed in the as-built reactor vessel before fuel loading. The staff finds the proposed ITAAC appropriate and complete.

4.4.4 Conclusions

The application meets the requirements of GDC 10 and 12 with respect to SAFDLs by providing analyses and test results demonstrating that normal operation, including the effects of AOOs, satisfy the fuel design criteria, provided that the conditions and limitations applicable to approved topical reports are satisfied. These topical reports describe the methods and assumptions used for the evaluation of the reactor thermal and hydraulic design. DCD Tier 2, Revision 9, Section 6.3, presents analyses related to core thermal and hydraulic design for emergency core cooling, and DCD Tier 2, Revision 9, Chapter 15, presents the transient and accident analyses. The corresponding sections of this report present the staff evaluation. DCD Tier 2, Revision 9, Appendix 4D, and Appendix 4A of this report specifically addresses GDC 12.

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

The staff reviewed DCD Tier 2, Revision 9, Section 4.5.1, in accordance with SRP Section 4.5.1. The CRD structural materials are acceptable if the relevant requirements of the following regulations are met:

- GDC 1, "Quality standards and records," and 10 CFR 50.55a(a)(1) require, in part, that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These quality standards shall be identified and evaluated to determine their adequacy to ensure a quality product, in keeping with the required safety function.

- GDC 14, "Reactor coolant pressure boundary," requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

- GDC 26 requires, in part, that one of the radioactivity control systems shall use control rods (preferably including a positive means for inserting the rods) and shall be capable of reliably controlling reactivity changes so that SAFDLs are not exceeded under conditions of normal operation, including AOOs.

Descriptive information on the FMCRD, as well as the entire CRD system, appears in DCD Tier 2, Revision 9, Section 4.6.1. As described below, the staff reviewed the structural materials aspects of the CRD, as presented in the DCD, in accordance with the guidelines in SRP Section 4.5.1 Revision 3.

4.5.1.1 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 4.5.1, describes the materials used to fabricate structural components of the CRD system. The DCD also provides information about the materials specifications, the fabrication and processing of austenitic stainless steel components, the contamination protection and cleaning of austenitic stainless steel, and items concerned with materials other than austenitic stainless steel.

The metallic structural components of the CRD mechanism are fabricated from four types of materials, which include 300 series stainless steel, nickel-chromium-iron (Ni-Cr-Fe) Alloy X-750, XM-19, and 17-4 PH materials. The primary pressure boundary components of the CRDs are the lower housing of the spool piece assembly, the flange of the outer tube assembly, and the mounting bolts. The applicant stated that all materials used in the CRD system are selected for their compatibility with the reactor coolant as described in Subarticles NB-2160 and NB-3120 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code).

Pressure-retaining materials comply with the ASME Code, Section III, which 10 CFR 50.55a incorporates by reference. DCD Tier 2, Revision 9, Table 5.2-4, includes the materials specifications for portions of the CRDs that are part of the RCPB and are fabricated from forged austenitic stainless steel (Type F316/F316L and Type F304/F304L).

The CRD system does not employ austenitic stainless steels strengthened by cold work. For incidental cold work introduced during fabrication and installation, special controls are used to limit the induced strain and hardness, and the bend radii are kept at a minimum value.

Stellite 3/Haynes 25 is used for rollers/pins at latch (outside), and Haynes 25 is used for the latch joint pin. A material equivalent to Stellite 6 is used in the guide shaft at the top of the ball spindle. Stellite 12 is used for the bushing at the top of the ball spindle and the bushing on the drive shaft. Stellite Star J-metal is used for the ball check valve. Non-cobalt hard surfacing alloys are used in guide rollers and guide pins. These components are located above and below the labyrinth seal and on the stop piston, ball screw stationary guide, piston head, and ball nut.

4.5.1.2 *Evaluation*

The staff reviewed and evaluated the information in DCD Tier 2, Revision 9, Section 4.5.1, to ensure that the materials specifications, fabrication, and process controls are in accordance with the criteria of SRP Section 4.5.1.

4.5.1.2.1 Materials Specifications

The staff reviewed DCD Tier 2, Revision 9, Section 4.5.1.1 to determine the suitability of the materials for this application. The DCD provides information on the specifications, types, grades, heat treatments, and properties used for the materials of the CRD components.

The CRD structural components that are part of the RCPB include the middle flange, the spool piece, and the mounting bolts. The middle flange and spool piece components are fabricated from austenitic stainless steel forgings (SA-336 or SA-182 F304/F304L/F316/F316L). The mounting bolts are SA-193, Grade B7. These materials comply with the requirements in the ASME Code, Sections II and III, and are acceptable for use in the ESBWR design.

The remaining components identified in DCD Tier 2, Revision 9, Section 4.5.1, are not RCPB materials. The DCD indicates that the properties of those components are equivalent to those given in Parts A, B, and D of Section II of the ASME Code or those included in RG 1.84, “Design, Fabrication and Materials Code Case Acceptability, ASME Section III, Revision 33” and are therefore acceptable for use in CRD components.

4.5.1.2.2 Austenitic Stainless Steel Components

The applicant indicated that all stainless steel materials are used in the solution heat-treated condition. For all welded components exposed to service temperatures exceeding 93 degrees C (200 degrees F), the carbon content in the austenitic stainless steel components is limited, not to exceed 0.020 percent. Limiting the carbon content in welded components experiencing service temperatures exceeding 93 degrees C (200 degrees F) to 0.020 percent or less is consistent with NUREG–0313, Revision 2, "Technical Report on Materials Selection Processing Guidelines for BWR Coolant Pressure Boundary Piping," which is consistent with SRP Section 4.5.1. The applicant indicated that significantly cold-worked 300 series austenitic stainless steels are not used. However, if minor forming and straightening are performed, the process will be controlled by limiting the material hardness, bend radius, or the amount of strain induced by the process. In RAI 4.5-31, the staff requested the applicant to provide the values of the ESBWR design special controls limits on hardness, 0.2-percent offset yield strength, and induced strain. The staff also requested the applicant to discuss the abrasive work controls for limiting cold working and the introduction of contaminants during abrasive processes. Finally, the staff requested the applicant to provide its response in a generic sense as it applies to the entire ESBWR design.

In response, the applicant stated the following:

> GEH applies special cold work controls to all stainless steel in the reactor system, defined as components inside containment continuously exposed to reactor water greater than 93 °C (200 °F). Bulk hardness of all stainless steels in the final fabricated condition (with the one exception noted in the response to RAI 4.5-12) is controlled to Rockwell B-90 for Types 304/304L and Rockwell B-92 for Types 316/316L. Cold forming and straightening strains are limited to 2.5 percent, or alternately, in the case of bars, plate, or pipe, a bend radius greater than 20 d or t (diameter or thickness). Additionally, for the major structural welds of core support structures and large internal components, polishing of the weld heat affected zones is required to remove surface cold work introduced by forming, machining, or grinding. Maximum yield strength is not controlled specifically, but the combination of solution heat treatment controls, hardness controls, and cold forming controls assure that, in all cases, the yield strength of stainless steels is far below 90,000 psi. Grinding is controlled by requiring ground areas to be polished to remove surface cold work introduced by grinding. Grinding media are controlled by requirements that processing materials shall be low in halogens, sulfur, and low melting point metals as well as thorough final cleaning of all ground surfaces. Additionally, it is required that grinding media be new, or previously used only on stainless steel or nickel alloys.

The maximum hardness limit as specified by the applicant is consistent with the acceptance criteria specified in SRP Section 4.5.1. The staff considers the applicant's special cold-work controls for all stainless steel components in the reactor system adequate to reduce the susceptibility of stainless steel materials to SCC resulting from cold working including grinding. Based on the applicant's response, RAI 4.5-31 is resolved. DCD Tier 2, Revision 9, Section 4.5.1.2.1, states that Section 4.5.2.2 discusses the degree of conformance to RG 1.44, "Control of the Use of Sensitized Stainless Steel." In Section 4.5.2.2, the applicant indicated that the ESBWR design complies with the intent of RG 1.44, which provides the acceptance criteria for testing, alloy compositions, welding, heat treatment, cleaning, and protecting austenitic stainless steels to avoid severe sensitization. In RAI 4.5-29, the staff requested the applicant to clarify its compliance with the guidelines of RG 1.44 because its use of the word "intent" does not make it clear whether the ESBWR design is consistent with all of the guidelines in RG 1.44. The staff also requested the applicant to specify the type of test it will use to detect susceptibility to intergranular attack in austenitic stainless steels in the ESBWR design in order to conform to the guidance in RG 1.44.

In response, the applicant clarified that it used the word "intent" in a general sense and indicated that sensitized stainless steel will not be used. The test used to detect susceptibility to intergranular attack is a modified version of American Society for Testing and Materials (ASTM) A 262, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Practice A, that more strictly defines rejectable ditching and that does not allow retest and acceptance under Practice E. The applicant also indicated that the ESBWR design will comply with RG 1.44. The staff finds this acceptable because the applicant will conform to the guidelines in RG 1.44 as listed in SRP Section 4.5.1. Based on the applicant's response, RAI 4.5-29 is resolved.

RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," provides the acceptance criteria for delta ferrite in austenitic stainless welds. These acceptance criteria address the recommended range of delta ferrite in stainless steel weld metal to avoid microfissuring in welds. The RG also includes a recommended procedure for ferrite measurement. DCD Tier 2, Revision 9, Section 4.5.2.2, indicates that the staff guidance in RG 1.31 will be followed, which is acceptable.

4.5.1.2.3 Other Materials

The DCD identifies that the bayonet coupling, latch and latch spring, and separation spring (non-RCPB components) are fabricated from Alloy X-750 in the annealed condition and aged 20 hours at 704 degrees C (1,300 degrees F). In RAI 4.5-28, the staff requested that the applicant discuss the relationship between the thermal and mechanical processing of Alloy X-750 components and their susceptibility to SCC.

In response, the applicant referred to its response to RAI 4.5-13 regarding Alloy X-750. In that response, the applicant indicated that the heat treatment for Alloy X-750 components, other than a spring on a shroud head bolt or a latch component in the steam dryer, is consistent with the Type 3 heat treatment of ASTM/ASME B/SB-637, "Specification for Precipitation-Hardening Nickel Alloy Bars, Forgings, and Forging Stock for High-Temperature Service" and the EPRI guidelines on X-750. The high-temperature anneal treatment in conjunction with a single step aging treatment is considered to provide optimum stress-corrosion resistance in X-750 in BWR applications. The applicant also indicated that, although it is believed that hardness in excess of Rockwell C40 (Rc40) can indicate elevated susceptibility to SCC, B/SB-637 Type 3 heat treatment specifies an Rc40 maximum hardness. Based on industry experience using X-750 in CRD components, the use of EPRI heat treatment guidelines, and the accessibility of these components for inspection and replacement if necessary, the staff finds the applicant's use of Alloy X-750 for CRD components acceptable. Based on the industry's experience reported in the applicant's response, RAI 4.5-28 is resolved.

The CRD ball spindle and ball nut are fabricated from 17-4 PH stainless steel in condition H-1075 (aged 4 hours at 579 degrees C [1,075 degrees F]). SRP Section 4.5.1 identifies 579 degrees C (1,075 degrees F) as an appropriate aging temperature for CRD components fabricated from 17-4 PH stainless steel. Therefore, the staff finds the applicant's heat treatment of 17-4 PH acceptable.

In the CRDs, cobalt-bearing and non-cobalt-bearing alloys are specified for wear and hard surfacing applications. Radiation buildup during plant operation can occur because of cobalt-60, which forms by neutron activation of cobalt-59. In RAI 4.5-27, the staff requested the applicant to discuss the basis for selection, operating experience with the materials selected, and use of cobalt-bearing and non-cobalt-bearing wear-resistant alloys in the ESBWR design.

In response, the applicant stated the following:

> Other than the cobalt bearing materials in the FMCRD noted in DCD Section 4.5.1, no cobalt bearing alloys are used in the ESBWR internals design. The components in the FMCRD are small bearings and other parts where maximum wear resistance is required. Because these materials are contained within the CRD, they are not directly activated because of being located far below the bottom of active fuel where neutron fluence is minimal. Release of cobalt to reactor water by general corrosion is very limited because the operating temperature inside the drive is substantially lower than reactor temperature, flow rates are low, and these cobalt base alloys have generally high corrosion resistance. The non-cobalt alloys used in wear and hard surfacing applications in the FMCRD components were selected specifically to minimize the use of cobalt base alloys. These alloys were qualified for the FMCRD application by extensive mockup testing for ABWR and have been in service in Kashiwazaki-Kariwa 7 since it started up in 1997. Any of these components are readily replaceable as part of routine CRD maintenance.

The staff finds the applicant's response acceptable, given that direct activation of cobalt-bearing alloys is unlikely and release of cobalt because of general corrosion is limited. Extensive mockup testing and a service history using these alloys for FMCRD components provide assurance of the capability of these materials to perform their intended function. Based on the above discussion in the applicant's response, RAI 4.5-27 is resolved.

4.5.1.2.4 Compatibility of Materials with the Reactor Coolant

The materials selected for use in the CRD system must be compatible with the reactor coolant, as described in Subarticles NB-2160 and NB-3120 of the ASME Code, Section III. The information in the DCD indicates that the RCPB materials used in the CRD system are compatible with the reactor coolant and, thus, comply with the ASME Code, Section III, Subarticles NB-2160 and NB-3120.

Furthermore, the materials selected for the CRD system are currently in use in nuclear power plants and have been proven to perform satisfactorily under the environmental conditions found in these plants. The staff finds the selected materials for this element of design to be acceptable because they perform satisfactorily under the expected environmental conditions.

4.5.1.2.5 Cleaning and Cleanliness Control

The staff's acceptance criteria for cleaning and cleanliness controls conform to RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The ESBWR design conforms to RG 1.37, with the exception of quality standard American National Standards Institute (ANSI) N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," 1973 referenced in RG 1.37. DCD Tier 2, Revision 9, Section 4.5.1.4 references

Part 2.2 of NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants," and RG 1.37. In RAI 4.5-30, the staff asked the applicant to specify the edition of NQA-1 that is applicable. The staff notes that the DCD references NQA-1-1983, in Chapter 17, but the applicable section related to the requirements for cleaning of fluid systems and associated components is located in NQA-2-1983, "Quality Assurance Requirements for Nuclear Facility Applications." The staff requested the applicant to provide clarification and state whether all positions of RG 1.37 are being met in a global context as it applies to the entire ESBWR design.

In response, the applicant stated the following:

> The ESBWR design commitment in DCD Table 1.9-22 will be changed to NQA-1-1983 and NQA-2-1983 in response to NRC review of DCD Chapter 17. All references in the DCD to NQA-1 and/or NQA-2 will be revised accordingly. The ESBWR design complies with RG 1.37 except as noted in DCD Table 1.9-21b. The NRC has accepted an alternate position as documented in Table 2-1 of DCD Reference 1.9-2 (GEH Nuclear Energy Quality Assurance Program Description, March 31, 1989, NEDO-11209-04a, Class I (non-proprietary) Revision 8). The alternate position is stated as follows: "Comply with the provisions of Regulatory Guide 1.37, March 16, 1973, including the requirements and recommendations in ANSI N45.2.1-1973, except as follows:
>
> "Section 5, sixth paragraph, recommends that local rusting on corrosion resistant alloys be removed by mechanical methods. This recommendation shall be interpreted to mean that local rusting may be removed mechanically, but that it does not preclude the use of other removal means. In addition, the ESBWR design complies with the cleaning requirements of ANSI N45.2.1-1980 and the packaging, shipping, receiving, storage and handling requirements of ANSI N45-2.2-1978 as referenced in DCD Table 1.9-22. Compliance is met by means of their incorporation into NQA-2-1983. DCD Section 4.5 will be revised in the next update to specify NQA-2-1983, Part 2.2 in Subsection 4.5.1.4 instead of NQA-1, Part 2.2."

The staff verified that the applicant made the above-cited changes to the DCD. Given that the staff has previously approved the use of NQA-1-1983, NQA-2-1983, and the applicant's alternative, the staff finds that the applicant meets the guidelines provided in RG 1.37, and its position is, therefore, acceptable. Based on the above discussion and the applicant's response, RAI 4.5-30 is resolved.

4.5.1.3 *Conclusions*

The staff finds the selection of materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls to be acceptable because they satisfy regulatory requirements or positions described above (for RCPB materials), or because they have been demonstrated to be acceptable based on appropriate materials selections and acceptable operating experience (for non-RCPB materials).

Based on the above, the staff concludes that the design of the CRD structural materials is acceptable and meets the requirements of GDC 1, 14, and 26, as well as 10 CFR 50.55a.

4.5.2 Reactor Internal Materials

The staff reviewed DCD Tier 2, Revision 9, Section 4.5.2 in accordance with SRP Section 4.5.2, Revision 3. The design, fabrication, and testing of the materials used in the reactor internals and core support structures are acceptable if they meet codes and standards commensurate with the safety functions to be performed. The acceptability of the materials will ensure that the relevant requirements of 10 CFR 50.55a and GDC 1 are met. The following specific acceptance criteria of SRP Section 4.5.2 are necessary to meet the requirements of 10 CFR 50.55a and GDC 1:

- Materials Specifications, Selection, and Heat Treatment

  For core support structures and reactor internals, ASME Code, Section III, Subarticle NG-2000, identifies and describes the permitted materials specification. ASME Code cases approved for use identify additional permitted materials and their applications, as described in RG 1.84.

  All materials used for reactor internals and core support structures must be compatible with the reactor coolant, as described in ASME Code, Section III, Subarticles NG-2160 and NG-3120. The tempering temperature of martensitic stainless steels should be specified to provide assurance that these materials will not deteriorate in service.

- Controls on Welding

  Methods and controls for core welding support structures and reactor internals must conform to ASME Code, Section III, Subarticle NG-4000. The welds must be examined and meet the acceptance criteria specified in ASME Code, Section III, Subarticle NG-5000.

- Nondestructive Examination

  Nondestructive examination (NDE) shall conform to the requirements of ASME Code, Section III, Subarticle NG-2500. The acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Subarticle NG-5300.

- Austenitic Stainless Steels

  SRP Section 5.2.3 Subsections II.2 and II.4.a, b, d, and e, provide the acceptance criteria for the reactor internal materials.

  RG 1.44 describes acceptance criteria for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, as well as for determining the degree of sensitization that occurs during welding. RG 1.31 describes acceptable criteria for ensuring the integrity of welds in stainless steel components.

- Other Materials

  All materials used for reactor internals and core support structures must be selected for their compatibility with the reactor coolant, as described in ASME Code, Section III, Subarticles NG-2160 and NG-3120. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are aging at 565 degrees C to 595 degrees C (1,050 degrees F to

1,100 degrees F) for type 17-4 PH and tempering at 565 degrees C (1,050 degrees F) for type 410 stainless steels. Other materials shall have similarly appropriate heat treatment and fabrication controls in accordance with strength and compatibility requirements.

4.5.2.1 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 4.5.2, describes the materials used to fabricate reactor internal and core support materials. Specifically, the DCD provides information about the materials specifications, controls on welding, NDE of wrought seamless tubular products, fabrication and processing of austenitic stainless steel components, and items concerned with materials other than austenitic stainless steel. Each of these topics is discussed below.

Materials Specifications

The DCD requires that all core support structures be fabricated from ASME-specified materials and designed in accordance with the criteria of ASME Code, Section III, Subsection NG. The other reactor internals are non-ASME Code, and they may be fabricated from ASTM or ASME specification materials or other equivalent specifications.

Controls on Welding

The DCD requires that core support structures be fabricated in accordance with the criteria of ASME Code, Section III, Subarticle NG-4000, and the examination and acceptance criteria included in Subarticle NG-5000. The reactor internals, other than the core support structures, meet the criteria of the industry standards (e.g., ASME or American Welding Society), as applicable. The qualification criteria of ASME Code, Section IX, are followed in the fabrication of core support structures. All welds are made with controlled weld heat input.

Nondestructive Examination of Wrought Seamless Tubular Products

The DCD requires that the stainless steel CRD housings, which are partially core support structures (inside the reactor vessel), serve as the reactor coolant boundary outside the reactor vessel. The CRD housing material is supplied in accordance with the criteria of ASME Code, Section III Class 1. The CRD housings are examined and tested in accordance with ASME Code, Section III, Subsection NB for the pressure boundary portion of the housing and in accordance with ASME Code, Section III, Subsection NG for the non-pressure boundary portion.

The peripheral fuel supports are supplied in accordance with ASME Code, Section III Subsection NG. The material is procured and examined according to ASME Code, Section III Subarticle NG-2500.

Wrought seamless tubular products for other reactor internal components are supplied in accordance with the applicable ASTM or ASME materials specifications. These specifications require a examination on each length of tubing or pipe.

Regulatory Guide Conformance for the Fabrication and Processing of Austenitic Stainless Steel

The DCD requires that significantly cold-worked stainless steels not be used in the reactor internals except for vanes in the steam dryers. Applying limits on hardness bend radii, and surface finish on ground surfaces, controls cold work. Furnace-sensitized material is not allowed. Electroslag welding is not applied for structural welds. The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 ferrite number (FN) minimum, 8.0 FN average, and 20 FN maximum. This ferrite content is considered adequate to prevent any microfissuring (hot cracking) in austenitic stainless steel welds in compliance with RG 1.31.

The limitation placed on the delta ferrite in austenitic stainless steel castings is a minimum value of 8 percent and a maximum value of 20 percent. The maximum limit is used for those castings designed for a 60-year life, such as the fuel support pieces, to limit the effects of thermal aging degradation. Proper solution annealing of the 300-series austenitic stainless steel is verified by testing in accordance with ASTM A-262 "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels". Welding of austenitic stainless steel parts is performed in accordance with ASME Code, Section IX (welding and brazing qualification), and Section II, Part C (welding rod electrode and filler metals).

All cleaning materials and process materials that contact stainless steel during manufacture and construction are controlled to prevent exposure to contaminants. Any inadvertent surface contamination is removed to avoid potential detrimental effects.

Special care is exercised to ensure removal of surface contaminants before any heating operation. Water quality for rinsing, flushing, and testing is controlled and monitored. The degree of cleanliness obtained by these procedures meets the criteria of RG 1.37.

Other Materials

The DCD specifies that hardenable martensitic stainless steel and precipitation hardening stainless steels are not to be used for the reactor internals. Materials, other than type 300 stainless steel, used in reactor internals are type or grade XM-19 stainless steel, niobium-modified Alloy 600 and N07750 (Alloy X-750), or equivalent. All niobium-modified Alloy 600 material is used in the solution-annealed condition and meets the criteria of ASME Code Case N-580-1. Alloy X-750 components are fabricated in the annealed and aged condition. In those areas that require maximum resistance to stress corrosion, the material is used in the high-temperature 1,093 degrees C (1,999.4 degrees F) annealed plus single aged condition.

Hard chromium plating surface is applied to austenitic stainless steel couplings. All materials used for reactor internals are selected for their compatibility with the reactor coolant as specified in ASME Code, Section III, Subarticle NG-3120.

4.5.2.2 *Evaluation*

The staff divided its evaluation of the reactor internals and core support materials in DCD Tier 2, Revision 9, Section 4.5.2, into five topics equivalent to those described in SRP Section 4.5.2. These topics include materials specifications, controls on welding, NDE, fabrication and processing of austenitic stainless steel components, and other materials and considerations.

4.5.2.2.1 Materials Specifications

DCD Tier 2, Table 4.5-1, does not identify many of the reactor internal components discussed in Section 2 of DCD Tier 1. In RAI 4.5-1, the staff requested the applicant to revise Table 4.5-1 in

DCD Tier 2 to include all core support structures and reactor internal components used in the ESBWR with corresponding materials specifications. The staff also recommended that Table 4.5-1 in DCD Tier 2 be revised to differentiate the core support structure components from the reactor internal components. In response, the applicant proposed a revision to Table 4.5-1 of DCD Tier 2 to include all reactor internal and core support components discussed in Section 2 of DCD Tier 1. The applicant replaced DCD Tier 2, Table 4.5-1, in the next update of the DCD. The staff finds that this proposed revision contains sufficient information on the materials specifications for all significant reactor internal and core support structure components and, therefore, is acceptable. Based on the applicant's response, RAI 4.5-1 is resolved.

DCD Tier 2, Section 4.5.2, contained no drawings of the core support structures or reactor internals. In RAI 4.5-2, the staff requested the applicant to provide the detailed drawings of all significant core support structures and reactor internal components, as well as assembly drawings to show how the core support structure components and reactor internal components are attached to each other and/or to the reactor vessel. The staff also suggested that DCD Tier 2, Section 4.5.2, include the drawings and diagrams. In response, the applicant provided reactor internals assembly drawings to supplement Figure 2.1.1-1 in DCD Tier 1. The drawing shows the assembly of the major core structures and internal components listed in the revised Table 4.5-1. In addition, the applicant provided conceptual drawings of the shroud, top guide, chimney, chimney partition, and core plate-to-shroud joints. Based on the applicant's response, RAI 4.5-2 is resolved.

In response to RAI 4.5-18, RAI 4.5-19, and RAI 4.5-20, the applicant also provided sketches of other reactor internal components. The applicant stated it would include the assembly drawings in Section 3.9.5 of DCD Tier 2 when it is next updated. The staff finds that the drawings clarify how reactor internals and core support structures are assembled and supported. Therefore, this issue is resolved. Based on the applicant's responses, RAI 4.5-18, RAI 4.5-19, and RAI 4.5-20 are resolved.

DCD Tier 2, Table 4.5-1 identifies cast austenitic stainless steel as a material that will be used in the ESBWR reactor internals. The staff noted that cast austenitic stainless steel is susceptible to a loss of fracture toughness because of thermal aging embrittlement, neutron irradiation embrittlement, and void swelling in the reactor vessel. The staff's concern was based on a letter from Christopher I. Grimes of the NRC to Douglas J. Walters of the Nuclear Energy Institute, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,'". In addition, ultrasonic examinations of cast austenitic stainless steel have not been reliable. In RAI 4.5-3, the staff requested the applicant to address the aforementioned degradation mechanism and its concern about the ability to inspect components made with cast austenitic stainless steel.

The applicant responded that the use of cast stainless steel components for the ESBWR internals is very limited and confined to components that are common to previous BWR designs. As such, these components have more than 35 years of BWR operating experience with no known problems or failures. The one core support application is the fuel support casting. This is a removable, replaceable piece of hardware on which the fuel bundles sit. In this location, the casting is well below the bottom of active fuel length and, as such, sees relatively low neutron dose compared to other core support structures such as the shroud and top guide. Neutron-induced void swelling does not occur because both the temperature and fluence are well below the nominal thresholds for this phenomenon in stainless steels (see "Irradiation Temperature Dependence of Void Formation in Type 304 Stainless Steel," by Sandusky et al.). The applicant stated that thermal aging is also not a concern. At the normal operating temperature for all

BWRs of 288 degrees C (550 degrees F), thermal aging of low-carbon stainless steel castings with less than 20-percent ferrite is barely measurable (NUREG/CR–5385, “Initial Assessment of the Mechanisms and Significance of Low-Temperature Embrittlement of Cast Stainless Steels in LWR Systems,” issued August 1990). To ensure that the potential for thermal aging is thoroughly limited, the applicant specified a maximum ferrite value of 20 percent for the castings.

The applicant stated further that the steam separator swirl generator castings and connector castings in the steam dryer are the only other castings in the ESBWR internals design. The swirler casting is a non-ASME Code, nonsafety-related functional component that sees essentially no neutron dose because of its location. The only structural demands on this casting are from directing the flow of steam and water and supporting the minor weight of the individual separator assembly to which the casting is welded. As with the fuel support casting, this component is unchanged from early BWR designs, except that the carbon content is now limited to the L-grade range (i.e., low carbon), and ferrite content is controlled to a range of 8 to 20 percent. The steam dryer castings are also non-ASME Code, nonsafety-related components that experience essentially no neutron dose. As with the castings mentioned above, they are low carbon with ferrite control. As such, the steam dryer castings are highly resistant to thermal aging and SCC. Because none of these castings is subject to ultrasonic testing, either in fabrication or in service, their ability to be inspected is not an issue.

The staff finds that, based on the applicant’s response, thermal aging embrittlement, neutron irradiation embrittlement, and neutron void swelling are not a concern for the cast austenitic stainless steel in the ESBWR environment. In addition, the ability of the subject material to be inspected is moot because the steam dryer is a non-ASME Code, nonsafety component that is not required to be inspected by ultrasonic testing. The staff finds that the applicant’s use of cast austenitic stainless steel is acceptable; therefore, this issue is resolved. Based on the applicant’s response, RAI 4.5-3 is resolved.

DCD Tier 2, Table 4.5-1 identifies that non-L-grade 304 and 316 stainless steels are used for the reactor vessel internals and core support structures. In RAI 4.5-5, the staff requested the applicant to justify the use of non-L-grade 304 and 316 stainless steels in light of the industry experience of intergranular stress-corrosion cracking (IGSCC) in 304 and 316 stainless steel in the BWR environment. The applicant responded that the carbon content is limited and should not exceed 0.02 percent in all welded wrought austenitic stainless steel components in the ESBWR that are exposed to reactor water at temperatures exceeding 93 degrees C (199.4 degrees F). The difference between 304 and 304L (or 316 and 316L) is in their respective mechanical strengths. The applicant proposed to add a footnote to DCD Tier 2, Table 4.5-1, requiring that the carbon content of all type or grade 304/304L or 316/316L used in the core support structures and reactor internal components be limited to a maximum of 0.02 percent. The applicant added this limitation to Table 4.5-1 in the next update of the DCD. The staff finds this explanation acceptable. Based on the applicant’s response, RAI 4.5-5 is resolved.

The staff notes that IGSCC has occurred in 304 and 316 stainless steel material in the BWR coolant environment as a result of sensitization. The high carbon content in 304 and 316 stainless steel contributes to this sensitization. According to the materials specifications of SA-240 in the ASME Code, Section II, the carbon content for 304/316 and 304L/316L stainless steel is limited to 0.08 percent and 0.03 percent, respectively. However, as the applicant stated above, the DCD limits the carbon content of 304, 304L, 316, and 316L stainless steels to 0.02 percent, which is lower than the ASME Code specifications for either stainless steel type.

The low-carbon-content requirement should minimize the potential for sensitization of 304 and 316 stainless steel, which in turn will minimize the potential for IGSCC. Therefore, the staff finds that the revised materials specifications for type 304 and 316 stainless steel in DCD Tier 2, Revision 9, Table 4.5-1, are acceptable.

In RAI 4.5-6, the staff requested the applicant to clarify the ASME Code and non-ASME Code components used in the reactor internals, identify the specific materials specification for each of the reactor internal components, and include this information in DCD Tier 2, Table 4.5-1. The applicant responded that those reactor internals with a core support function are fabricated and certified to ASME Code, Section III Subsection NG. All other ESBWR internal components are considered "internal structures," consistent with Subsection NG terminology. For these components, materials may be procured that meet either ASTM or ASME Code, Section II, standards or equivalents. The individual ASTM and corresponding ASME materials specifications are essentially identical (e.g., ASTM A-240, type 316L plate is identical to ASME SA-240, type 316L). The applicant proposed to revise DCD Tier 2, Table 4.5-1 in the next update to identify the materials specifications of reactor internals and core support structures. The staff finds that the proposed revision to DCD Tier 2, Table 4.5-1 provides specific ASME or ASTM materials specifications and clarifies the difference between ASME Code and non-ASME Code material. Therefore, based on the applicant's response, RAI 4.5-6 is resolved.

In RAI 4.5-7, the staff requested the applicant to (1) discuss the operating experience (i.e., degradation) of the non-ASME Code materials used in the reactor internals in the current BWR fleet, (2) demonstrate that the non-ASME Code material will provide the strength, resistance to corrosion, and fracture toughness necessary to maintain the safe operation of the ESBWR, (3) discuss whether the non-ASME Code components are designed for and analyzed with the same loading combinations, in accordance with the ASME Code, Section III, as that used for the ASME Code components, and (4) clarify whether the non-ASME Code components are considered as safety or nonsafety category components.

The applicant responded to RAI 4.5-7 with the following:

(1) As discussed in the response to RAI 4.5-6, the materials used for internal structures are identical in chemistry and properties to their ASME Code counterparts. Consequently, there is no distinction in behavior in BWR service between the ASME Code core support structures and other internal structures.

(2) Strength, corrosion resistance, and toughness of the internal structure materials are equivalent to that of their ASME Code counterparts.

(3) Internal structures are designed and analyzed using Article NG-3000 of ASME Section III, Subsection NG, as a guideline. Loading combinations are the same as those specified for core support structures. Stresses and fatigue usage factors will meet the limits specified in Subsection NG.

(4) Internal structures may be safety-related or nonsafety-related, depending on their function. The standby liquid control line is an example of a safety-related internal structure. Non-safety internal structures include such components as the steam separators and steam dryer.

The staff finds that the applicant's response is satisfactory. Based on the applicant's response, RAI 4.5-7 is resolved.

In RAI 4.5-8, the staff requested the applicant to discuss which industry standards will be used for material selection, fabrication, construction, design, testing, and inspection for the non-ASME Code components. The applicant responded that the non-ASME Code materials used for the internal structures are identical in chemistry and properties to the ASME Code materials used for the core support structures. Consequently, there is no distinction in behavior in BWR service between the ASME Code core support structures and non-ASME Code reactor internal components. The applicant stated further that strength, corrosion resistance, and toughness of the non-ASME Code internal structure materials are equivalent to those of their ASME Code counterparts. Non-ASME Code internal structures are designed and analyzed using ASME Code, Section III Subsection NG-3000 as a guideline. Loading combinations are the same as those specified for core support structures. Stresses and fatigue usage factors will satisfy the limits specified in ASME Code, Section III, Subarticle NG. Internal structures may be safety-related or nonsafety-related, depending on their function. The standby liquid control line is an example of a safety-related internal structure. Non-safety internal structures include components such as the steam separators and steam dryer. The applicant stated further that material selection and fabrication for the non-ASME Code components are consistent with the ASME Code. Welding procedures and welders are qualified to ASME Code, Section IX. Inspection methods are consistent with ASME Code, Section V, and the acceptance criteria that follow ASME Code, Section III Subsection NG.

The staff finds that the non-ASME Code components are designed and analyzed using ASME Code, Section III, as a guide. The structural performance of the non-ASME Code components in terms of strength, corrosion resistance, and toughness is equivalent to that of the ASME Code components. Therefore, based on the applicant's response, RAI 4.5-8 is resolved.

4.5.2.2.2 Controls on Welding

SRP Section 4.5.2 specifies that the methods and controls for core support structure and reactor internal welds must be performed in accordance with ASME Code, Section III, Division 1, Subarticle NG-4000, and the welds must be examined and meet acceptance criteria as specified in Subarticle NG-5000. However, DCD Tier 2, Section 4.5.2.2 discussed the welding of the reactor internals without referring to the relevant ASME Code sections. In RAI 4.5-9, the staff asked the applicant to identify the ASME Code sections relevant to core support structure and reactor internal components that require welding and to describe the welding technique and procedures. In addition, the staff requested the applicant to clarify the intent of DCD Tier 2, Section 4.5.2.2, which does not explicitly mention welding.

The applicant responded that "fabrication," as used in DCD Tier 2, Section 4.5.2.2, is intended to encompass all fabrication processes, including welding as defined in ASME Code, Section III, Subarticle NCA-9000. For core support structures, the components are required to be built and certified in full compliance with ASME Code, Section III Subsection NG. Therefore, compliance with Subarticles NG-4000 and NG-5000 is implicit, and all welding will be performed and inspected accordingly. The applicant did not consider it necessary to explicitly refer to select portions of Subsection NG in the DCD because full compliance with Subsection NG in its entirety is required. For the non-ASME Code internal components, welding qualification according to ASME Code, Section IX, is required. Welding practices and inspections are generally consistent with ASME Code, Section III, NG-4000 and NG-5000. Most of the core support structures and reactor internals require some welding for assembly. The main exceptions are the fuel supports that rest on the core plate, which are machined from forgings or castings. Welding processes will be those commonly applied to stainless steels and nickel alloys, such as shielded metal arc welding, gas tungsten arc welding, submerged arc welding, and gas metal arc welding. Both manual and automatic processes will be applied. The specific welding techniques and procedures cannot be defined at this time because such details depend on the facility contracted to do the fabrication work.

The staff finds that the applicant has clarified the welding processes and referenced the relevant ASME Code sections for the core support structures and reactor internals. Based on the applicant's response, RAI 4.5-9 is resolved.

4.5.2.2.3 Nondestructive Examination

SRP Section 4.5.2.II.3, Draft Revision 3, issued April 1996, specifies that the acceptance criteria for NDE shall be in accordance with the requirements of ASME Code, Section III Subarticle NG-5300. However, DCD Tier 2, Section 4.5.2.3, does not specify the acceptance criteria for NDE. In RAI 4.5-10, the staff requested the applicant to include the acceptance criteria for NDE in DCD Tier 2, Section 4.5.2.3. The applicant responded that, for core support structures, full compliance with ASME Code, Section III, Subsection NG, is understood and so stated. Similarly, for the reactor internal components that have a pressure-retaining function, full compliance with ASME Code, Section III, Subsection NB, is required and so stated. These subsections contain acceptance criteria for NDE. Therefore, the applicant did not consider it essential for the DCD to explicitly reference individual articles of the ASME Code, such as NB/NG-5300. The staff finds that the applicant has clarified the ASME Code sections relevant to the acceptance criteria for NDE of the reactor internal and core support structure components. Based on the applicant's response, RAI 4.5-10 is resolved.

DCD Tier 2, Section 4.5.2.3 discusses the NDE of CRD housings and peripheral fuel supports but is silent on other reactor internal components. SRP Section 4.5.2.I.3, Draft Revision 3, issued April 1996, recommends that each product form in the reactor internals and core support structures be examined. In RAI 4.5-11, the staff requested the applicant to justify why NDE is not required for product forms other than CRD housings and peripheral fuel supports. The staff also requested that the applicant identify which specific tubular products will be hydrostatically tested. The applicant responded that it would revise DCD Tier 2, Section 4.5.2.3, to reflect the expanded scope of Draft Revision 3 to SRP Section 4.5.2.I.3. Examination of core support structure materials and welds will be in full compliance with ASME Code, Section III Subsection NG. In addition, pressure-retaining components and welds will be inspected in full compliance with Subsection NB. The applicant proposed to revise DCD Tier 2, Section 4.5, in the next update as described in the following sections.

Materials for core support structures will fully conform and be certified to ASME Code, Section III Subsection NG. Subarticle NG-2500 specifies the examination of materials (examination methods and acceptance criteria). Subarticle NG-5000 provides examination methods and acceptance criteria for core support structure weld edge preparations and welds. Tubular products that are pressure boundary components (CRD and in-core housings) will be examined according to ASME Code, Section III, Subarticle NB-2500, and associated pressure-retaining welds will be examined according to Subarticle NB-5000. For non-ASME Code reactor internal structures and associated welds, examinations are established based on relevant design and analysis information and will follow guidance from ASME Code, Section III, Subarticles NG-2500 and NG-5000, respectively.

The staff finds that the proposed revision to DCD Tier 2, Section 4.5.2.3, satisfies SRP Section 4.5.2, and, therefore, is acceptable. Based on the applicant's response, RAI 4.5-11 is resolved.

In RAI 4.5-14, the staff requested the applicant to discuss the pre-service inspection and inservice inspection program of all core support structure and reactor internal components. For each component, the staff requested that the discussion include specific examination techniques, frequency of the inspection, acceptance criteria, the area/coverage of the inspection, and the industry codes/requirements used. The staff also requested the applicant to provide a list of components that will not be inspected during the pre-service or in-service inspection activities and explain why the inspection is not needed.

The applicant responded that the pre-service and in-service inspections of core support structures and internal components are a COL holder (i.e., Licensee) issue. However, the applicant stated that visual examination of the core support structures will be performed during plant outages as required by ASME Code, Section XI, Table IWB-2500-1, Item B13.40. The frequency of the examinations will be conducted as identified in Subarticle IWB-2400 of the ASME Code, Section XI. The examination personnel shall be qualified in accordance with Subarticle IWA-2300. The ASME Code has no requirements for pre-service and in-service inspections of reactor internal components that are non-ASME Code components. These components include the chimney, chimney partitions and chimney restraints, chimney head and steam separator assembly, chimney head bolts, steam dryer assembly, feedwater spargers, standby liquid control piping and distribution headers, in-core guide tubes, in-core guide tube restraints, guide rods, and drain pipes.

The applicant stated further that during the fabrication of core support structures all material is examined as required by ASME Code, Section III, Subarticle NG-2500. For the examination of non-ASME Code internal components, the ASME Code is used as a guideline. A liquid penetrant examination is required on the weld preparation surfaces before welding and on all machined surfaces. The extent of NDE of welds is determined by the weld quality and fatigue factors (ASME Code, Section III Table NG-3352-1) applied to the weld joints in the design analysis. All welds, materials, and subassemblies not accessible for inspection in the completed assembly are inspected for quality and cleanliness before the last activity that results in their inaccessibility.

A visual examination of the completed components that meets the requirements of ASME Code, Section XI, Subarticle IWA-2210, performed in the shop, serves as a “preservice visual inspection.” The same rigorous quality and cleanliness requirements are applied to the installation of the reactor internals in the field.

The staff finds that the general pre-service and in-service inspection of the reactor internal components and core support structures follows the ASME Code. Therefore, based on the applicant’s response, RAI 4.5-14 is resolved.

The BWR Vessel and Internals Project (BWRVIP) has published many guidelines related to the inspection of reactor internals. The NRC has approved some of the BWRVIP reports. In RAI 4.5-15, the staff requested the applicant to discuss which BWRVIP guidance and reports will be used. The applicant responded that the BWRVIP guidelines were written for maintenance, inspection, and repair of currently operating BWRs and do not address new plant construction. Consequently, these guidelines are not specifically used to establish ESBWR requirements. However, ESBWR materials selection and controls are generally consistent with the EPRI “Advanced Light Water Reactor Utilities Requirements Document.” The staff finds this explanation acceptable, and therefore, based on the applicant’s response, RAI 4.5-15 is resolved.

4.5.2.3 *Fabrication and Processing of Austenitic Stainless Steel Components*

DCD Tier 2, Revision 9, Section 4.5.2.4, states that significantly cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. In RAI 4.5-12, the staff requested the applicant to justify the use of cold-worked materials in vanes, considering the adverse impact of the cold work on the microstructure of the material and the susceptibility of cold-worked materials to SCC. The applicant responded that some degree of cold working is necessary to form the steam dryer vane shape. This design is essentially unchanged from the earlier BWRs. Thus, over 35 years of operating experience with this design has accumulated, and no failures of vanes have been observed. The material has been updated to current low carbon standards, and maximum hardness is controlled to a level well below the threshold for SCC. Because the only function of the vanes is to direct steam flow, these parts experience virtually no sustained tensile stress.

The applicant stated further that, even if SCC were to occur, there is virtually no potential to create a loose part because the vane banks are contained between perforated plate assemblies. The staff finds this explanation to be acceptable. Therefore, based on the applicant's response, RAI 4.5-12 is resolved.

DCD Tier 2, Section 4.5.2.5 identifies Alloy X-750 as a material that will be used in the reactor internals. However, Alloy X-750 materials are susceptible to IGSCC because of equalized and aged heat treatment conditions (BWRVIP-41, "BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," EPRI 1012137). In RAI 4.5-13, the staff requested the applicant to (1) identify the reactor internal components fabricated from Alloy X-750, (2) provide information on the aging heat treatment process of Alloy X-750 (i.e., aging temperature and holding time), (3) justify how this aging process will help to minimize SCC, (4) provide the optimal hardness value that is required to minimize the susceptibility to SCC, and (5) discuss why Alloy X-750 is identified in DCD Tier 2, Section 4.5.2.5, but not in DCD Tier 2, Table 4.5-1.

The applicant responded that, other than the CRD components identified in DCD Tier 2, Section 4.5.1, the use of Alloy X-750 in ESBWR internal components is very limited. The only application positively identified at this time is a coil spring on the shroud head bolt (a nonsafety-related component) and possibly a latch component in the steam dryer. Age hardening of Alloy X-750 will consist of a single-step treatment. The coil spring will be age hardened for 16 hours at 732 degrees C (1,350 degrees F). Any other shapes of Alloy X-750 will be age hardened at 704 degrees C (1,299.2 degrees F) for 20-21 hours. The annealing process performed before the aging treatment produces most of the improved SCC resistance of Alloy X-750. ESBWR Alloy X-750 components will be annealed at 1,080-1,108 degrees C (1,975-2,025 degrees F). This is the heat treatment condition developed for improved BWR jet pump beam performance in the early 1980s. It is consistent with the Type 3 heat treatment of the ASTM/ASME B/SB-637 material and the EPRI guidelines on Alloy X-750. The heat treatment, in conjunction with a single-step aging treatment, is considered to provide optimum stress-corrosion resistance to Alloy X-750 in BWR applications. Hardness has not been identified as a control parameter for SCC resistance, except that it is known that hardness in excess of Rc40 can indicate elevated susceptibility. Type 3 heat treatment of ASTM/ASME B/SB-637 material specifies a Rockwell C40 maximum hardness. The applicant also committed to revising DCD Tier 2, Table 4.5-1 to include the known components fabricated with Alloy X-750. The staff finds the applicant's response acceptable because all Alloy X-750 components in the reactor internal components are nonsafety-related, and any potential cracking of these components will not adversely impact safe operation of the reactor vessel. In addition, the Alloy X-750 components in the ESBWR will have undergone a high-temperature anneal treatment and a single-step aging treatment to improve their corrosion-resistance performance. Therefore, based on the applicant's response, RAI 4.5-13 is resolved.

In RAI 4.5-16, the staff asked the applicant to discuss the maintenance program for the bolts and threaded fasteners used in the core support structures and reactor internals to ensure their structural integrity and to prevent them from becoming loose parts in the reactor coolant system. The applicant responded that cracking of bolts and fasteners in the core support structure and reactor internal components has not been an issue in operating BWR plants. Positive locking mechanisms are used for bolting applications (e.g., nuts are tack welded in place to prevent them from coming loose), and visual inspections are performed during installation. Austenitic stainless steel bolts and nuts of types 304, 304L, 316, and 316L have been generally used in the past. However, in newer plants, including the ESBWR, nitrogen-strengthened austenitic stainless steel, grade XM-19, material is being used for high-load bolted joints. Because the ASME Code, Section XI, has no requirement for in-service inspection of the bolts and because of the favorable BWR operating experience, there are no formal ESBWR maintenance and inspection requirements for bolts and threaded fasteners inside the RPV. The staff finds this explanation acceptable. Based on the applicant's response, RAI 4.5-16 is resolved.

In RAI 4.5-17, the staff requested from applicant to identify the ASME Code requirements for material selection, inspection, design, fabrication, and construction of the chimney, chimney partitions, and chimney head; to describe the fabrication, assembly, and installation of the chimney, chimney partitions, and chimney head; and to discuss whether a mockup test of the chimney assembly in a reactor vessel environment has been conducted to verify the structural integrity of the chimney assembly.

The applicant responded that ASME Code, Section III, Subsection NG, was used as a guideline for the material, design, fabrication, and inspection of the chimney, chimney partitions, and chimney head. These components are classified as internal structures and do not require an ASME nuclear code stamp. As discussed above, they are non-ASME Code components. The chimney partition assembly consists of a grid of square structures, each of which encompasses 16 fuel assemblies and a bottom and a top ring.

The bottom ring rests on, and is pinned and bolted to, the bottom flange of the cylindrical chimney shell. The top ring of the assembly is supported against the inside of the chimney shell. The chimney assembly is bolted to the top guide and laterally supported by eight chimney restraints at the top. As discussed in DCD Tier 2, Revision 9, Appendix 3L, an air and water two-phase flow vibration test of both a 1/6-scale and a 1/12-scale model of a single chimney cell was performed. The results of the scale testing were extrapolated by a two-phase flow analysis to determine the characteristics of the pressure fluctuations acting on the partition wall of a full-size cell in steam-water conditions. The stress analysis showed an adequate margin against the allowable vibration peak stress amplitude based on the test results. The staff finds that, even though the chimney is a non-ASME Code component, ASME Code, Section III, Subsection NG, is used as a guide for the material selection, design, fabrication, and inspection of chimney components. Based on the applicant's response, RAI 4.5-17 is resolved.

The core shroud supports in the operating BWR fleet are supported from and attached to the bottom of the reactor. However, for the ESBWR, the core shroud is attached and supported at the side wall of the reactor vessel, which may produce a bending moment on the vessel wall.

The staff was concerned that the shroud supports may not sustain the loads as calculated in the structural analysis because the vessel wall may not be as rigid as assumed in the analysis. In RAI 4.5-18, the staff requested the applicant to discuss whether the design of the core shroud supports considered the potential bending of the reactor vessel wall and whether the stress analysis of the reactor vessel shell considered the bending moment generated by the core shroud supports. In addition, the staff noted that the core shroud supports use niobium-modified Inconel 600 Alloy, which is susceptible to SCC. The staff requested the applicant to justify the selection of this material and to provide the drawings and design details, including the location and installation of the core shroud supports.

The applicant responded that shroud supports that are attached directly to the reactor vessel wall have been used in vessels built by Combustion Engineering (e.g., Plant Hatch). Analyses and experience have proven that the bending stresses produced by the cantilever shroud support design in these vessels are acceptable.

The bending moment from the shroud support will be included in the ESBWR design documentation containing the reactor vessel stress analysis. Since the bending moment from the ESBWR shroud support is smaller than in the aforementioned vessels because of a smaller gap between the shroud and the vessel wall, excessive bending stresses are not expected.

The applicant stated further that the core support material is Ni-Cr-Fe Alloy 600 with niobium added. ASME Code Case N-580-1 permits the use of niobium. Niobium-modified Ni-Cr-Fe Alloy 600 has been successfully used in the ABWRs, and tests have shown that it is highly resistant to SCC in a BWR environment. The staff finds it acceptable to include the bending moment of the core support structures in the reactor vessel shell analysis in the design documentation. In addition, the subject core shroud support design in currently operating BWRs has not adversely affected any reactor vessel walls. Therefore, the staff finds that the shroud support design is acceptable. Based on the applicant’s response, RAI 4.5-18 is resolved.

In RAI 4.5-19, the staff requested the applicant to (1) provide assembly drawings of the CRD housing and stub tube to show how they are attached to each other and to the bottom of the vessel and (2) discuss weld joint details, welding processes, post weld heat treatments, materials to be used, and the fabrication sequence to be used to prevent sensitization of the stainless steel material (e.g., operating experience at Oyster Creek). In response, the applicant provided a schematic drawing of the reactor vessel CRD penetrations, which shows that the stub tubes are welded to the Ni-Cr-Fe cladding in the bottom head of the reactor vessel. The stub tube material is niobium-modified Ni-Cr-Fe Alloy 600 in accordance with ASME Code Case N-580-1. Welding of the joints between the stub tubes and the bottom head, and between the CRD housings and the stub tubes, is performed with a process using nickel Alloy 82 filler material, according to ASME SFA-5.14, Grade ER NiCr-3 (or use of Alloy 182; according to SFA-5.11, Grade ER NiCrFe-3 is not permitted). The final post weld heat treatment of the vessel is performed after the NiCrFe stub tubes are welded into the bottom head. This type of stub tube connection and material has been successfully used in the recent ABWRs. The staff notes that the welds using nickel Alloy 82 filler material in pressurized-water reactors have experienced primary SCC. However, in the BWR environment, Alloy 82 weld metal is considered to be acceptable for use because of its resistance to IGSCC (NUREG–0313, Revision 2). In addition, the drawings provided by the applicant clarify how the CRD penetrations are attached to the stub tubes on that basis. Based on the applicant’s response, RAI 4.5-19 is resolved.

Section 2.1.1 of DCD Tier 1 states that a lattice work of clamps, tie bars, and spacers provides lateral support and rigidity to the in-core guide tubes. In RAI 4.5-20, the staff requested the applicant to provide assembly drawings of the lateral support components and in-core guide tubes to show how the lateral support components are interconnected and how the in-core guide tubes are attached to the reactor vessel. The staff also requested that the applicant include the drawings in DCD Tier 2, Section 4.5.2 identify materials used for the lateral support components and in-core guide tubes, and identify the number of penetrations.

The applicant responded that the lower ends of the in-core guide tubes are welded to the in-core housings in the bottom of the reactor vessel. The top ends extend through holes in the core plate, which provides lateral support. DCD Tier 2, Section 3.9.5, will include conceptual drawings illustrating the interconnections between the in-core guide tubes' lateral supports and their attachments to the lower portion of the shroud, and the connections between the guide tubes and the core plate. The revised DCD Tier 2, Table 4.5-1, contained in the applicant's response to RAI 4.5-1, which identified the material of the lateral supports and the in-core guide tubes. The reactor vessel bottom head has a total of 88 in-core penetrations. DCD Tier 2, Figures 7.2-6 and 7.2-7 show the locations of the penetrations within the core. The applicant revised DCD Tier 2, Section 3.9, to include this information. The staff finds that the applicant has provided drawings to clarify the configuration of the supports for the in-core guide tubes and associated lateral supports. Therefore, the staff finds that this issue is resolved. Based on the applicant's response, RAI 4.5-20 is resolved.

DCD Tier 1, Section 2.1, states that special controls on material fabrication processes will be exercised when austenitic stainless steel is used for the construction of reactor internals. The staff issued In RAI 4.5-22, to request the applicant to describe the special controls that are used for material fabrication. The applicant responded that the controls will be contained in the detailed purchase specifications used to procure materials and fabricate components. Consequently, the full level of detail is not yet in place for the ESBWR. However, when these documents are prepared, the content will be very similar to existing specifications for ABWRs. For preparation of individual equipment documents, guidance will be taken from "Materials and Processes Controls," a top-level ESBWR materials and process document. The general practice is to have a materials specification that is used in conjunction with a fabrication specification for individual groups of equipment. For stainless steel materials, a number of controls are placed on the supplier that is more detailed than the basic ASTM and/or ASME requirements. In addition to the 0.02-percent maximum carbon limitation that will be included in the revised DCD Tier 2, Table 4.5-1, these equipment requirements documents will include the following among the controls generally applied:

- Limitations on cobalt content (varies depending on proximity to the core)

- Detailed controls on heat treatment time/temperature and quenching

- For nuclear grade 304/316 material, confirmatory test of yield strength at 288 degrees C (550 degrees F)

- Control of maximum hardness

- Sensitization test (modified ASTM A-262, Practice A)

- Intergranular attack control

- Limitations/controls on weld repairs
- Cleaning, marking, and packaging controls

Fabrication of stainless steel components will be controlled using detailed fabrication specifications that include the following:

a. Controls on hardness (e.g., control of mechanical cutting methods, machining controls, grinding controls, controls on cold bending, forming and straightening, and limitations on both bulk and surface final hardness)

b. Controls on thermal processes (e.g., thermal cutting methods and heat input, hot forming and bending, and specific controls of induction bending)

c. Welding controls (e.g., joint configurations, fit-up and gap, alignment, permitted processes, heat input control, back-purge and flux controls, allowed filler metals, ferrite control) and measurement method, weld metal control and storage, and RG 1.71, "Welder Qualification for Areas of Limited Accessibility," regarding restricted access qualification

d. Control of repairs, including allowed weld repairs

e. Detailed NDE requirements

f. Cleaning and cleanliness controls, including control of miscellaneous process materials;

g. Traceability of material, marking, and packaging for shipment

The staff finds that the applicant has satisfactorily identified the controls on the material fabrication processes. Therefore, based on the applicant's response, RAI 4.5-22 is resolved.

4.5.2.4 *Other Materials and Considerations*

In RAI 4.5-21, the staff requested the applicant to clarify whether a hydrogen water chemistry program will be implemented in the ESBWR to mitigate SCC. The applicant responded that the materials were selected and process controls were identified without taking credit for the application of hydrogen water chemistry. The ESBWR design calls for the reactor internal components to be capable of operating for the design life of the ESBWR without experiencing SCC failures. The licensee may choose to adopt hydrogen water chemistry primarily for added corrosion resistance (no deleterious effects on the selected materials). The ESBWR design does incorporate features (e.g., injection taps) that facilitate installation of the hydrogen water chemistry system either before or after initial startup. The staff finds that it is appropriate for the ESBWR design to include features to facilitate future installation of the hydrogen water chemistry system. Based on the applicant's response, RAI 4.5-21 is resolved.

Operating BWRs have experienced cracking of the feedwater spargers (NUREG–0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," November 1980). In RAI 4.5-23, the staff requested the applicant to describe design features, fabrication processes, and water chemistry to minimize or prevent cracking in feedwater nozzles and spargers in the ESBWR. The staff also requested the applicant to discuss the inspection program for the feedwater spargers.

The applicant responded that cracking of the feedwater spargers in some of the earlier BWRs was caused by three mechanisms: (1) high-cycle thermal fatigue caused by subcooled water leaking through the loose fit between the feedwater nozzle and the thermal sleeve, (2) subcooled water shedding from the subcooled thermal sleeve which periodically cooled the nozzle, and (3) thermal stratification in the feedwater sparger during low flow. In the ESBWR, the feedwater sparger, thermal sleeve, and vessel nozzle are welded together, thus eliminating the leakage flow of subcooled water. To prevent the reactor vessel nozzle from being exposed to cold water shedding from the thermal sleeve, licensees use a double thermal sleeve of a tuning fork design. The subcooled feedwater flows through the inner sleeve that is welded to the sparger. The concentric outer sleeve protects the vessel nozzle from being exposed to the cold water periodically shedding from the outer surface of the inner sleeve. The tuning fork design mitigates the thermal stresses between the austenitic stainless steel thermal sleeve and the low-alloy vessel nozzle. The ESBWR feedwater sparger has a row of spray nozzles mounted at the top of the sparger pipes so that the sparger will always be filled with water from the feedwater piping system, with minimal mixing with the warmer reactor vessel water. This sparger design helps to minimize thermal stratification within the sparger and piping during low flow conditions. Recent BWR product lines, as well as retrofit designs installed in the Monticello and Tsuruga-1 nuclear power plants in the early 1980s, have successfully used this sparger thermal sleeve design.

In reviewing the applicant's response to RAI 4.5-14, the staff finds that the applicant has adequately addressed the staff's concern about the potential for feedwater sparger cracking by specifying a design that has been used successfully in operating BWRs. Based on the applicant's response, RAI 4.5-23 is resolved.

In RAI 4.5-24, the staff requested the applicant to describe the programs that will be used to prevent and manage metallic loose parts in the reactor vessel during fabrication/assembly, maintenance, normal operation, and refueling activities. The applicant responded that fabrication and installation of the reactor vessel and the reactor internals are performed in accordance with a quality program that meets the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. This includes implementation of a cleanliness program. Before plant operation, the vessel and attached piping will be flushed to remove debris that may have collected during construction. The licensee will implement loose part controls during service and maintenance. The staff is satisfied with the applicant's response because the DCD includes a quality program that conforms to the requirements of Appendix B to 10 CFR Part 50 to prevent loose parts. Based on the applicant's response, RAI 4.5-24 is resolved.

In RAI 4.5-25, the staff requested the applicant to discuss the likelihood that the following degradation mechanisms will affect the core support structures and reactor internal components: ductile and brittle fractures, fatigue failures, distortion failures, wear failures, erosion (cavitation and liquid impingement), corrosion (pitting, leaching, galvanic, and intergranular), creep, corrosion fatigue, hydrogen-damage failures, embrittlement (neutron irradiation and thermal), and SCC. The staff also requested that the applicant discuss the steps taken to minimize or prevent these degradation mechanisms.

In a letter dated June 16, 2006, the applicant responded that the ESBWR is an evolutionary design that incorporates many aspects of prior BWR designs. In particular, the operating environment to which internal components are exposed is essentially identical. Therefore, more than 30 years of operating experience can be used to determine which degradation mechanisms may be active in the ESBWR. On that basis, the applicant addressed the aforementioned degradation mechanisms as follows:

- Ductile and brittle fractures: Use of ASME Code design rules ensure that there is no risk of ductile failures, even under upset conditions. Stainless steels and nickel alloys are not embrittled by fabrication processes, thermal aging, or exposure to BWR water. Although neutron irradiation decreases ductility, even at the highest exposure levels for reactor internals, significant residual toughness is retained.

- Fatigue failures: Fatigue failures have been very limited in BWR internals with a few exceptions. Historically, some fatigue failures have occurred in jet pump components, but jet pumps are not used in the ESBWR design. The other component that has experienced fatigue issues in operating BWRs is the steam dryer. The potential for fatigue failures in the ESBWR steam dryer is being addressed by implementation of a highly fatigue-resistant design based on extensive finite element and computerized fluid dynamic modeling, along with scale model testing.

- Distortion failures: Only one distortion failure has been observed in operating BWR internals. A series of steam dryers were fabricated with thin (0.3 cm [0.125-in.]) end hood plates, which became distorted by a pressure pulse generated by rapid MSIV closure. This problem was corrected by replacement with thicker hood material. The ESBWR steam dryer end hood plates are thicker than those used in some existing BWRs.

- Wear failures: Other than the CRDs and control rods, there are no moving parts in the ESBWR reactor internals. Wear has been considered by choosing hard-facing or wear-resistant alloys for moving components subject to wear. All the moving components that would potentially be subject to wear are routinely removable and replaceable.

- Erosion (e.g., cavitation and liquid impingement): This degradation phenomenon has not been observed in the internals of operating BWRs and is not expected in the ESBWR. Stainless steels are very resistant to erosion because of their high chromium content.

- Corrosion (e.g., pitting, leaching, galvanic, and intergranular): Stainless steels and nickel alloys have not been observed to experience corrosion phenomena in the BWR environment, which uses very pure deionized water.

- Creep: Stainless steels and nickel alloys do not experience creep at the maximum operating temperature of the ESBWR.

- Corrosion fatigue: A corrosion-fatigue interaction has not been observed in BWR internal components. The fatigue failures noted above are thought to have resulted from cyclic loading without any apparent or significant environmental factor. In any case, design improvements to eliminate the potential for fatigue failures in the ESBWR have addressed this concern.

- Hydrogen-damage failures: Hydrogen-driven failure mechanisms such as hydriding are not active in the BWR environment. Stainless steels and nickel alloys are not susceptible to hydrogen embrittlement or hydriding under the thermodynamic conditions in BWR water, even for a plant operating with hydrogen water chemistry.

- Embrittlement (neutron irradiation and thermal): Stainless steels and nickel alloys are not subject to thermal embrittlement at the ESBWR operating temperature (288 degrees C [550.4 degrees F]). Stainless steel does experience loss of ductility and toughness with neutron irradiation. This loss becomes significant at cumulative irradiation dose exceeding about 1x1021 n/cm2 (E > than 1 million electron volts [MeV], where E = energy). However, only certain areas of the reactor internals receive neutron doses exceeding this level, and even at the maximum dose for reactor internals, a significant degree of toughness is maintained. Operating BWRs achieve similar dose levels in reactor internals and no embrittlement failures have been observed, even in plants where there is frequent seismic activity.

- Stress-corrosion cracking: The design of the ESBWR addresses the potential for SCC of reactor internals by (1) use of only solution-annealed, low-carbon stainless steels and nickel alloys modified for high SCC resistance, (2) strict control of fabrication and installation processes, and (3) application of polishing to remove surface cold work in the weld heat-affected zones of the major structural welds in the large internals. These measures are expected to greatly reduce the potential for SCC of internals in the ESBWR relative to the currently operating BWRs. Routine inservice inspections will monitor the condition of the internals and be capable of detecting degradation by SCC in the unlikely event that it occurs.

The staff finds that the applicant has satisfactorily addressed the potential for degradation mechanisms by using appropriate material selection, fabrication, installation, and inspection of the core support structure and reactor internal components. On the basis of the above evaluation, the staff concludes that the reactor internal and core support structure components of the ESBWR design satisfy the acceptance criteria of SRP Section 4.5.2. The ESBWR design also satisfies RG 1.31 for the control of ferrite content in stainless steel weld metal, RG 1.37 for the cleanliness and quality of the fluid system to minimize corrosion of the austenitic stainless steel and loose parts, RG 1.44 for the control of the use of sensitized stainless steel, and RG 1.84 for the use of NRC-approved ASME Code cases. Therefore, the ESBWR design satisfies the relevant requirements of 10 CFR 50.55a and GDC 1. Based on the applicant’s response, RAI 4.5-25 is resolved.

4.5.2.5 *Conclusions*

On the basis of the information submitted, the staff concludes that the ESBWR design of the reactor internals and core support materials satisfies the relevant requirements of 10 CFR 50.55a and GDC 1, and therefore, is acceptable. This conclusion is based on the fact that the ESBWR reactor vessel internals and core support structures satisfy ASME Code, Section III; RGs 1.31, 1.37, 1.44, and 1.84; and SRP Section 4.5.2.

4.6 <u>Control Rod Drive System</u>

The CRD system controls changes in core reactivity during power operation by movement and positioning of the neutron-absorbing control rods within the core in response to control signals from the RC&IS and rapid control rod insertion (scram) in response to manual or automatic signals from the RPS.

4.6.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 4.6 in accordance with the regulatory guidance for the review of Control Rod Drive System, including adherence to applicable general design criteria (GDC) discussed in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (hereafter referred to as the SRP), Section 4.6, Draft Revision 2, issued June 1996. The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any generic issues (GI), bulletins (BL), generic letters (GL), or technically significant acceptance criteria (except Appendix 4B, Interim Criteria and Guidance for the reactivity initiated accidents) beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 2 of SRP Section 4.6, issued in June 1996, is acceptable for this review.

The staff's review covers the functional performance of the CRD system to confirm that the system can affect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents. Acceptance criteria are based on the following:

- GDC 4, "Environmental and dynamic effects design bases," as related to the environmental conditions caused by high- or moderate-energy pipe breaks during normal plant operation as well as postulated accidents

- GDC 23, "Protection system failure modes," as related to a failure of this system placing the reactor into a safe state

- GDC 25, as related to the functional design of redundant reactivity systems to ensure that SAFDLs are not exceeded for a malfunction of any reactivity control system

- GDC 26 as related to the capability of the reactivity control system to regulate the rate of reactivity changes resulting from normal operations and AOOs

- GDC 27 as related to the combined capability of reactivity control systems and the ECCS to reliably control reactivity changes to ensure the capability to cool the core under accident conditions

- GDC 28 as related to postulated reactivity accidents

- GDC 29, "Protection against anticipated operational occurrences," as related to functioning under AOOs

- 10 CFR 50.62(c)(3), the ATWS rule, related to diversity of the alternate rod injection system and redundancy of scram air header exhaust valves

SRP Section 4.6, Draft Revision 2 contains specific review procedures and acceptance criteria.

4.6.2 Summary of Technical Information

ESBWR DCD Tier 2, Revision 9, Section 4.6.1.2, describes the CRD system functions as follows:

- Controls change in-core reactivity by positioning neutron-absorbing control rods within the core in response to control signals from the RC&IS;

- Provides movement and positioning of control rods in increments to enable optimized power control and core power shape in response to control signals from the RC&IS;

- Provides the ability to position large groups of rods simultaneously in response to control signals from the RC&IS;

- Provides rapid control rod insertion (scram) in response to manual or automatic signals from the RPS so that no fuel damage results from any plant AOO;

- In conjunction with the RC&IS, provides automatic electric motor-driven insertion of the control rods simultaneously with hydraulic scram initiation, which provides an additional, diverse means of fully inserting a control rod;

- Supplies rod status and rod position data for rod pattern control, performance monitoring, operator display, and scram time testing by the RC&IS;

- In conjunction with the RC&IS, prevents undesirable rod pattern or rod motions by imposing rod motion blocks to protect the fuel;

- In conjunction with the RC&IS, reduces the probability of a rod drop accident by detecting rod separation and imposing rod motion block;

- In response to signals from the DPS, provides rapid control rod insertion (scram) and alternate rod insertion, an alternate means of actuating hydraulic scram, should an ATWS occur;

- In conjunction with the RC&IS, provides for SCRRI and select rod input (SRI);

- Prevents rod ejection from the core as the result of a break in the drive mechanism, pressure boundary, or a failure of the attached scram line by means of a passive brake mechanism for the FMCRD motor, and a scram line inlet check valve;

- Supplies high-pressure makeup water to the reactor when the normal makeup supply system (feedwater) is unable to prevent the reactor water level from falling below the normal water-level range;

- Supplies purge water for the reactor water cleanup (RWCU)/shutdown cooling (SDC) system pumps; and,

- Provides a continuous flow of water to the nuclear boiler system to keep the reactor water-level reference leg instrument lines filled.

The CRD system consists of three major elements:

(1) Electro-hydraulic FMCRD mechanisms;
(2) Control rod drive hydraulic system (CRDHS); and
(3) HCU assemblies.

Fine Motion Control Rod Drive

The fine motion capability is achieved with a ball-nut and ball-screw arrangement driven by an electric motor. The ball-nut is keyed to the guide tube to prevent its rotation and traverses axially as the ball-screw rotates. A hollow piston rests on the ball-nut, and upward motion of the ball-nut drives this piston and the coupled control rod into the core. The weight of the control rod keeps the hollow piston and the ball-nut in contact during withdrawal. The electric motor-driven ball-nut and ball-screw assembly is capable of positioning the drive at a nominal 36.5-millimeter (1.44-inch) increments.

Control Rod Drive Hydraulic System

The CRD system provides electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during abnormal operating conditions. High-pressure water stored in the individual HCUs provides the hydraulic power required for scram. Each HCU contains a nitrogen-water accumulator charged to high pressure and the necessary valves and components to scram two FMCRDs. Additionally, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs. The CRDHS supplies clean, demineralized water that is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs during normal operation. The CRDHS is also the source of pressurized water for purging the RWCU/SDC system pumps and the nuclear boiler system reactor water-level reference leg instrument lines. Additionally, the CRDHS provides high-pressure makeup water to the reactor during events in which the feedwater system is unable to maintain reactor water level. This makeup water is supplied to the reactor via a bypass line off the CRD pump discharge header that connects to the feedwater inlet piping via the RWCU/SDC return piping.

Hydraulic Control Unit

Each HCU furnishes pressurized water for hydraulic scram, on signal from the RPS, to drive two CRD units. Additionally, each HCU provides the capability to adjust purge flow to the drives. A test port is provided on the HCU for connection to a portable test station to allow controlled venting of the scram insert line to test the FMCRD ball check valve during plant shutdown. The nitrogen gas bottle provides a source of readily available high-pressure, high-discharge flow rate of nitrogen to the accumulator. The accumulator provides the stored energy necessary to obtain the required high-pressure, high-flow-rate discharge of water to the two associated FMCRDs. The accumulator has a floating piston with nitrogen on one side and water on the other side. The HCU also includes the scram solenoid pilot valve, scram valves, check valves, and restricting orifice.

4.6.3 Staff Evaluation

ESBWR DCD Tier 2, Revision 9, Section 4.6.1.1.1, specifies the safety design bases of the CRD system as follows:

- The design shall provide for the rapid control rod insertion (scram) so that no fuel damage results from any AOO.

- The design shall include positioning devices, each of which individually supports and positions a control rod.

- Each positioning device shall be capable of holding the control rod in position and preventing it from inadvertently withdrawing outward during any non-accident, accident, post-accident, or seismic condition.

- Each positioning device shall be capable of detecting the separation of the control rod from the drive mechanism to prevent a rod drop accident.

- Each positioning device shall provide a means to prevent or limit the rate of control rod ejection from the core resulting from a break in the drive mechanism pressure boundary. This is to prevent fuel damage resulting from rapid insertion of reactivity.

- The design provides for isolation capability, which terminates high pressure make up water, high pressure CRD, to ensure containment pressure remains within design limits.

The staff's review of the functional design of the ESBWR CRD system confirmed that it satisfies the above safety design bases and the regulatory criteria in Section 4.6.1 of this report. The staff's review of the functional design of the FMCRD system confirmed that the design has the following capabilities to satisfy the various reactivity control conditions for all modes of plant operations:

- The capability to operate in full-power mode throughout plant life

- The capability to vary power level from full power to hot shutdown and have power distributions within acceptable limits at any power level

- The capability to shut down the reactor to mitigate the effects of postulated events, which is discussed in Chapter 15 of this report

The ESBWR design incorporates electric-hydraulic FMCRDs that will provide electric fine rod motion during normal operation and hydraulic pressure for scram insertion. A ball-nut and spindle arrangement driven by the electric motor provides fine motion during normal operation. In response to a scram signal, the control rods will be inserted hydraulically by the stored energy in the scram accumulator, similar to the means of insertion in the currently operating BWR CRDs.

A scram signal is also given simultaneously to insert the FMCRDs electrically via the FMCRD motor drive. This diversity, which includes both hydraulic and electric methods of scramming, provides a high degree of assurance of rod insertion on demand.

The FMCRD is designed to control reactivity during power operation. Automatic rod insertion will control reactivity in the event of fast transients.

If the reactor cannot be shut down with the control rods, the operator can actuate the SLCS (if not automatically started), which injects a solution of sodium pentaborate into the primary system. Section 9.3.5 of this report addresses the evaluation of the functional design of the SLCS.

Section 15.5 of this report discusses compliance with the ATWS rule, 10 CFR 50.62. Section 3.6 of this report discusses compliance with GDC 4 requirements that the control rod drive system (CRDS) be designed to perform its safety-related functions and not be compromised by adverse environmental conditions caused by high- or moderate-energy pipe breaks.

The FMCRD will control reactivity in the core by moving control rods interspersed throughout the core. These rods will control the reactor's overall power level and will provide the principal means of quickly and safely shutting down the reactor.

The staff issued RAI 4.6-7, to request submittal of the failure modes and effects analysis (FMEA) for the FMCRD. In response, the applicant proposed that the FMEA for the ABWR FMCRD system is applicable to the ESBWR because the ABWR and ESBWR FMCRD systems are similar (except for a few items). ABWR DCD Tier 2, Revision 4, Appendix 15B provides the FMEA submitted for the ABWR. The text and descriptive material in Sections 15B.2.1, 15B.2.2, 15B.2.3, and 15B.2.4 are applicable to the ESBWR, with the exception that the FMCRD stepping motor of the ABWR design is replaced with the induction motor/magnetically coupled FMCRD design of the ESBWR. The staff reviewed the submitted material and concluded that the ABWR FMEA are unchanged by this difference and so is appropriate and applicable for the ESBWR FMEA analysis. The staff accepts this response. Therefore, based on the applicant's response, RAI 4.6-7 is resolved.

The single-failure analysis of the FMCRD and HCU components indicates that the system design is satisfactory. A supply pump (with a spare pump on standby) will provide the HCUs with water from the condensate and feedwater system or the condensate storage tank to supply CRD purge water and to supply the purge water to the RWCU pumps. The supply pump also will provide water to a scram accumulator in each HCU to maintain the desired water inventory. When necessary, the accumulator will force water into the drive system to scram the control rods connected to that HCU. The volume of water in the scram accumulator will be sufficient to scram two rods. A single failure in an HCU may result in the failure of two control rods. Section 4.3 of this report discusses the impact of this feature on shutdown margin.

The FMCRD is designed to permit periodic functional testing during power operation with the capability to independently test individual scram channels and the motion of individual control rods. The FMCRD is designed so that failure of all electrical power or instrument air will cause the control rods to scram, thereby protecting the reactor. This feature meets the requirements of GDC 23.

Preoperational tests of the CRDHS will be conducted to verify the capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week.

After each refueling shutdown, control rods will be tested for compliance with scram time criteria from the fully withdrawn position. Section 14.2 of this report presents the staff's evaluation of the preoperational and startup tests.

The FMCRD is designed to control reactivity under normal operating conditions and during AOOs and infrequent events. The safety analyses discussed in DCD Chapter 15 demonstrate this capability. The CRD system also will be capable of holding the core subcritical under cold shutdown conditions. The SLCS will be capable of bringing the reactor subcritical under cold-down conditions if the control rods cannot be inserted. These protection and reactivity control systems, taken together, satisfy the requirements of GDC 26, 27, and 29 pertaining to reactivity control system redundancy and capability, combined reactivity control system capability and protection against AOOs and infrequent events.

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns will be selected to achieve optimum core performance and low individual rod worth. The RC&IS will reduce the chances of withdrawal other than by the preselected rod withdrawal pattern. The reactor plant control system function will assist the operator with an effective backup control rod monitoring routine that enforces adherence to established control rod procedures for startup, shutdown, and low-power-level operations. A malfunction in these systems could result in either a local or global reactivity change. Chapter 15 of this report includes analysis of accident scenarios such as control rod withdrawal error. As part of that review, the staff evaluated the categorization of these reactivity events, their acceptance criteria, and compliance with GDC 25, as discussed in Section 4.2 and Chapter 15 of this report, specifically regarding RAI 4.2-6. The staff reviewed the compliance of the CRD system with GDC 25. Based on the applicant's response, RAI 4.2-6 is resolved in Chapter 15.

The safety concerns associated with a pipe break, described in NUREG–0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," issued August 1981, are not applicable to the ESBWR. The ESBWR design does not include scram discharge volume piping. The water displaced by the CRD during the scram will be routed to the RPV.

The high-pressure makeup mode of operation initiates automatically on receipt of a low-water Level 2 signal. The CRDHS supplies high-pressure makeup water to the reactor vessel (about 3,785 litres per minute (1,000 gallons per minute) with both pumps running simultaneously) through the RWCU/SDC. The flow is then routed through the feedwater system sparger for delivery to the reactor. At high reactor water Level 8, the high-pressure makeup flow control valve closes to stop flow to the reactor to prevent flooding of the main steamlines. Since the ESBWR has no high-pressure core makeup system, the high-pressure core makeup mode of operation is an important feature of the FMCRD system.

The CRD pumps are tripped by coincident low-water levels in two of the three GDCS pools to prevent containment pressurization.

In RAI 4.6-10, the staff requested the applicant to identify the portions of the CRD system that are safety-related and to describe how the safety-related portions of the system are isolated from the nonessential portion of the system.

In response, Applicant identified the following safety-related CRD system equipment:

- FMCRDs, including the following parts:
  - Primary pressure components
  - Hollow piston
  - Labyrinth seal
  - Latches
  - Guide tube
  - Brake (passive holding function)
  - Check valve
  - Check valve retainers

  - Internal anti-shootout (includes outer tube, outer tube to middle flange weld, and middle flange)
  - Parts that couple the brake with the hollow piston
  - Separation switches
  - Anti-rotation device

- HCU (scram circuit only)

- Scram insert piping

- Scram charging header pressure instrumentation

- High-pressure makeup piping at the connection to the RWCU/SDC system (including the check valve and injection valve)

According to the applicant, the CRD System is arranged in a manner that separates the safety-related equipment from the nonsafety-related portions of the system. The FMCRDs are mounted to the reactor vessel bottom head inside the primary containment. The HCUs are housed in four dedicated rooms located directly outside of the primary containment at the basemat elevation of the reactor building. These rooms are arranged around the periphery of the primary containment wall. Each HCU room serves the FMCRD associated with one quadrant of the reactor core. The HCUs are connected to the FMCRDs by the scram insert piping that penetrates the primary containment wall.

The balance of the nonsafety-related hydraulic system equipment (pumps, valves, filters, etc.) is physically separated from the HCUs and housed at a different elevation in the reactor building. It is connected to the HCUs by three nonsafety-related piping headers: the FMCRD purge water header, HCU charging water header, and scram air header. As shown in DCD Tier 2, Revision 9, Figure 4.6-8, these headers are classified as seismic Category II so that they will maintain structural integrity during a seismic event and not degrade the functioning of the HCUs.

The high pressure makeup piping at the connection to RWCU/SDC is classified as safety-related seismic Category I piping to provide interface compatibility with the safety-related seismic Category I piping of the RWCU/SDC.

As described above, the safety systems are adequately separated from the nonsafety system and hence RAI 4.6-10 is resolved.

ESBWR DCD Tier 2, Revision 9, Section 4.6.2.1.3, describes design features aimed at precluding excess reactivity events (e.g., rod ejection and rod drop events). The control rod mechanical design incorporates a brake system and ball check valve, which reduces the chances of rapid rod ejection. The ball check valve is classified as safety-related because it actuates to close the scram inlet port by reverse flow under system pressure, fluid flow, and temperature conditions caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod ejection. This engineered safeguard will protect against a high-reactivity insertion rate from a potential control rod ejection. Normal rod movement and the rod withdrawal rate will be limited through the FMCRD.

Applicant adopted an internal CRD housing support to replace the support structure of beams, hanger rods, grids, and support bars used in current BWR designs.

This system will use the outer tube of the drive to provide support. This tube will be welded to the drive middle flange and will be attached by a bayonet lock to the guide tube base. The guide tube, supported by the housing extension, will prevent downward movement of the drive in the event of housing failure. The CRD housing support is designed to prevent ejection of a CRD and attached control rod.

The FMCRD is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate safety related switches will be provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball-nut. Actuation of either of these switches will cause an immediate rod block and will initiate an alarm in the control room, thereby reducing the chances of a rod drop accident. Because of the design features described, the ESBWR control rod design does not include a velocity limiter.

Based on these design features, the applicant believes that the ESBWR design incorporates sufficient safeguards to negate its susceptibility to excess reactivity events. Initially, the DCD did not include design requirements or a CRDA analysis. The staff had concerns that several scenarios might lead to an excess reactivity event and that each case would require scenario specific analysis to ensure that it was beyond design basis. If any scenario was credible, acceptance criteria (e.g., coolability, radiological consequences) would need to be developed and an acceptable accident analysis performed to demonstrate that these criteria were satisfied. The inclusion of this family of accidents may involve changes to the proposed ESBWR TSs (e.g., limiting conditions for operation (LCOs), engineered safety features actuation system (ESFAS) setpoints) and the DCD (e.g., Sections 4.2 and 4.6 and Chapter 15). RAI 4.6-23 requested further information on the ESBWR design features and the probability and consequences of each accident scenario leading to an excess reactivity event. The staff reviewed the control rod drop event frequency estimates provided by applicant in response to RAI 4.6-23. The design and testing of the control rod and CRD mechanism include a number of diverse and redundant features for preventing a rod drop event, which is an indicator of high reliability in the design. Based on its review of key design and operational features and the fault-tree analysis provided by the applicant, the staff concludes that the applicant has presented a reasonable estimate of the rod drop frequency. However, the staff has also considered the applicant’s control rod drop event frequency evaluation provided in response to RAI 4.6-23 S01. Based on the potential consequences of an unrestricted reactivity excursion and to ensure compliance with GDC 28, the staff determined that the applicant must demonstrate RCPB integrity for the ESBWR design and acceptable radiological consequences for the CRDA irrespective of the probability of CRDA. SRP Section 4.2, Appendix B, offers more detailed regulatory criteria and guidance. The staff required this regulatory position to be updated in DCD Tier 1, Section 2.2.2; DCD Tier 2, Section 4.6; and DCD Tier 2, Section 15.4.6. In response, the applicant met the SRP Section 4.2, Appendix B criteria by analyzing the CRDA. Section 15.4.7 of this report includes the staff evaluation of the CRDA. Based on the results of the staff evaluation of DCD Tier 2, Section 15.4.6 and the applicant’s response, RAI 4.6-23 is resolved.

ESBWR DCD Tier 2, Revision 9, Section 4.6.1.2.2, describes the support of the fuel assemblies and the core support plate. DCD Tier 2, Revision 9, Section 4.1.2.1.2 states, “Each guide tube, with its orificed fuel support, bears the weight of four assemblies and is supported on a CRD penetration nozzle in the bottom head of the reactor vessel.” The staff issued RAI 4.6-26 to request additional information concerning the design margin between the control rod guide tube flange elevation and core support plate elevation. Specifically, the staff requested that the applicant address: (1) thermal expansion and contraction of the reactor vessel and (2) differential growth between the reactor vessel and the control rod guide tube.

In response, the applicant stated that there is no contradiction between DCD Tier 2, Subsections 4.6.1.2.2 and 4.1.2.1.2. Subsection 4.1.2.1.2 describes the reactor configuration in its normal state. In this condition the weld between the CRD housing and the CRD penetration nozzle in the reactor bottom head carries the full weight of the four assemblies, the orificed fuel support, the control rod guide tube, and the FMCRD. Subsection 4.6.1.2.2 describes the rod ejection condition in which the weld between the CRD housing and CRD penetration nozzle fails completely. In this case the control rod guide tube drops down a distance equal to the normal gap until its flange at the top engages with the core plate. Based on the above response, the staff considers RAI 4.6-26 to be resolved.

ESBWR DCD Tier 2, Revision 9, Section 4.6.1.2, describes the CRD system functions, including the "ability to position large groups of rods simultaneously." With the ability to move multiple control rods simultaneously comes the possibility to inadvertently move multiple rods. This inadvertent withdrawal would introduce a core wide power transient that would be more global than the traditional localized rod withdrawal error event. The inclusion of this accident may involve changes to the proposed ESBWR TSs (e.g., LCOs, ESFAS setpoints) and the DCD (e.g., Sections 4.2 and 4.6 and Chapter 15). The staff issued RAI 4.6-27 to request more information on the core and plant systems' response to a rod withdrawal error event involving large groups of rods. The applicant responded that mitigation of spurious rod movement by one or more rods is provided by RC&IS functions. A rod withdrawal error at power is protected by the RWM and automated thermal limit monitor subsystems of the RC&IS that terminate any spurious rod movement of one or more rods before violation of the OLMCPR. Any disagreement between the two RC&IS channels initiates a rod block (unless one is bypassed). Any one channel can signal rod block. Section 15.3 of this report addresses rod withdrawal error during power. Based on the applicant's response and the results of the review of Section 15.3, the staff considers RAI 4.6-27 is resolved.

DCD Tier 2, Revision 9, Section 4.6.1.2, describes the CRD system functions, including the provision of SCRRI. An inadvertent control rod run-in would result in a redistribution of core power and potentially an approach to a fuel design limit. The inclusion of this accident may involve changes to the proposed ESBWR TSs (e.g., LCOs, ESFAS setpoints) and the DCD (e.g., Sections 4.2 and 4.6 and Chapter 15). The staff issued RAI 4.6-28 to request additional information on the core and plant systems' response to an inadvertent control rod run-in event. The applicant responded by stating that SCRRI is an automatic function of the RC&IS and CRD system in the ESBWR design. The CRD system also provides FMCRD run-in. This automatic ATWS mitigation feature uses the FMCRDs to run in all the control rods in an emergency. The applicant enhanced the SCRRI function in DCD Tier 2, Revision 3, to include simultaneous hydraulic insertion of rods, known as SRI. (See DCD Tier 2, Revision 9, Section 7.1.5.4.10.) With the addition of SRI, an inadvertent SCRRI/SRI actuation as described below does not challenge core thermal limits. The quick response of the SRI rods reduces core power without creating an axial power transient that could potentially challenge fuel thermal limits. DCD Tier 2, Revision 9, Figure 15.2-4 shows the response to a generator load rejection with turbine bypass. Except for the slight pressure transient at the beginning of the event, the response is very similar to an inadvertent SCRRI/SRI. As shown, the SRI quickly reduces the core power. Although the radial power distribution does change, the core power reduction is significant enough to ensure that thermal limits are not challenged. Analysis shows that an inadvertent run-in of a single FMCRD would not challenge thermal limits. In a follow-up the staff issued RAI

4.6-28 S01 to request additional information regarding the instances of SCRRI and/or SRI failure that may affect core symmetry in power distribution. In response, the applicant satisfactorily addressed functions of SRI and SCRRI, partial SCRRI failure, and partial SRI insertion disturbing core symmetry and introducing instabilities. Staff review of this issue is found in Section 15.3 of this report. Based on the applicant's response, RAI 4.6-28 S01 is resolved.

ESBWR DCD Tier 2, Revision 9, Section 4.6.1.2.1, describes the spring-loaded latches on the hollow piston that engages slots in the guide tube. These latches support the control rod and hollow piston in the inserted position following a scram. The staff issued RAI 4.6-29 to request additional information regarding possible latch failure and the significant power peaking and loss of shutdown margin. In their response the applicant stated that the holding function of these latches will be tested and confirmed via the continuous full-in position indicator light as part of the scram testing defined in ITAAC 12 in DCD Tier 1, Table 2.2.2-7. The applicant also included details of the slot locations in the guide tube wall. Based on the applicant's response and the defined testing in ITAAC 12, RAI 4.6-29 is resolved.

ESBWR DCD Tier 2, Section 4.6.1.2.2, states, "Each FMCRD provides two position detectors, one for each control system channel, in the form of signal detectors directly coupled to the motor shaft through gearing." This section goes on to state, "This configuration provides continuous detection of rod position during normal operation." The staff issued RAI 4.6-30 to request additional information regarding the accuracy of the position indication. In response, the applicant stated that the signal detectors sense the number of rotations of the FMCRD ball screw and translate that information into an analog signal corresponding to control rod position. The cited position accuracy comprises the variation in braking distance and the accuracy of position detection. The applicant stated that this system configuration is identical to that in the ABWR design and is based on European FMCRD designs that have many years of reliable operating experience. Based on the applicant's response regarding the role of the ball screw and that it has been applied to the ABWR, RAI 4.6-30 is resolved.

ESBWR DCD Tier 2, Revision 9, Section 4.6.1.2.2, describes the FMCRD components. This section discusses the spring-loaded control rod separation mechanism. The staff issued RAI 4.6-31 to request additional information regarding its concerns that over time, irradiation-induced spring relaxation might impact the ability of this mechanism to perform its safety-related function. In response, the applicant stated that these mechanisms would not be exposed to significant neutron fluence because of the shielding provided by several meters of water in the reactor vessel between the core plate and the vessel bottom head. The staff agrees with this explanation. Based on the fact that there are several meters of water between the bottom of active core and the bottom of the vessel which provides significant neutron attenuation, RAI 4.6-31 is resolved.

DCD Tier 2, Section 4.6.1.2.2 also discusses the FMCRD electromechanical brake and states that a "braking torque of 49 N-m (minimum) and the magnetic coupling torque between the motor and the drive shaft are sufficient to prevent control rod ejection in the event of failure in the pressure retaining parts of the drive mechanism." The staff issued RAI 4.6-32 to request calculational information of the minimum torque required to prevent rod injection. In response, the applicant provided details of this calculation which, when based on conservative inputs, shows that the calculated torque on the ball screw (resulting from loading associated with a break in the scram line) remains below the 49 Newton-meter (N-m) (433.7 in.-pounds) design breaking torque requirement. The staff finds this calculation acceptable. The minimum holding torque of 49 N-m (433.7 in.-pounds) will be verified as part of ITAAC 15 in DCD Tier 1, Table

2.2.2-7. Based on the applicant's response and the minimum value of 49 N-m (433.7 in.-pounds), RAI 4.6-32 is resolved.

DCD Tier 2, Section 4.6.3.5, states, "A test of the scram times at each refueling outage is sufficient to identify any significant lengthening of the scram times." ESBWR TS Surveillance Requirement 3.1.4.2 requires routine (e.g., every 200 days) sampling of scram times for a representative set of control rods. Based on recent experience with channel bow, the staff believes that routine scram tests are necessary to detect the onset of control rod interference resulting from channel bow and to ensure control rod operability and satisfaction of scram time requirements. The staff issued RAI 4.6-33 to request additional information regarding the planned testing to detect control rod interference. In response, the applicant stated that it did not intend to remove the routine scram testing. A subsequent revision to this DCD section clarified the requirement for routine testing. Based on the applicant's response and the revision of the DCD regarding testing, RAI 4.6-33 is resolved.

DCD Tier 2, Revision 9, Section 4.6.1.2.6, describes a rod withdrawal block signal generated because of rod-gang misalignment. The staff issued RAI 4.6-34 to request additional information on the allowable gang misalignment (before rod block), the accuracy of measuring the misalignment, and whether any safety analysis or LCO accounts for this misalignment. The applicant's response included the explanation that the rod action and position indication A and B monitor the gang rod position and issue a rod block by sending appropriate rod block signals to the logic of the rod server processing channels in the remote communication cabinets if the gang misalignment exceeds a predetermined value. Section 15.3 of this report discusses rod gang misalignment in more detail. The staff reviewed the supplemental information and based on the applicant's response, RAI 4.6-34 is resolved.

DCD Tier 2, Section 4.6.3.5, describes the surveillance test for the high-pressure makeup mode but does not state the frequency for this surveillance. The staff issued RAI 4.6-35 to request the frequency of the surveillance reported in Section 4.6.3.5. In response, the applicant stated that it intended a test frequency comparable to that for a safety-related, motor-driven, high-pressure ECCS pump. A subsequent revision to this DCD section included the surveillance tests and frequencies; therefore, based on the applicant's DCD revision, RAI 4.6-35 is resolved.

Standard TSs require certain surveillance tests following maintenance and before declaring a system operable. DCD Tier 2, Section 4.6.3.5, includes no such requirements. The staff issued RAI 4.6-36 to request DCD modifications that reflect post-surveillance testing. In response, the applicant stated that scram time tests were required on each affected control rod following maintenance. A subsequent revision to this DCD section reflected this requirement. Therefore, based on the DCD modification, RAI 4.6-36 is resolved.

The ESBWR CRD system design represents a departure from that of the currently operating BWR fleet. The staff issued RAI 4.6-37 to request discussion of the CRD operating experience in systems similar to ESBWR. The staff noted that the proposed CRD differs significantly from the US operating fleet CRDs. In response regarding reactor operating experience with similar CRD system designs, the applicant described the commercial service of a similar design in the Japanese ABWRs. In approximately 20 reactor-years of service, these reactors have experienced no anomaly indicating a fundamental or serious design issue. With respect to manufacturing and testing experience, FMCRDs and HCUs have been manufactured to design specification both for the Japanese ABWRs and for an ongoing Taiwanese project and have successfully passed performance testing requirements. The operating and manufacturing experience supplied by the applicant provides reasonable assurance that the ESBWR CRD system can be manufactured to satisfy all design requirements. Based on the applicant's response regarding ABWR operating experience, RAI 4.6-37 is resolved. The Tier 1 ITAAC will ensure that the CRD system installed at each ESBWR site satisfies these requirements.

Section 7.8.3 of this report includes the staff evaluation of the requirements pertaining to 10 CFR 50.62, the alternate rod injection system, and redundant scram air header exhaust valves.

4.6.4 Conclusions

The staff concludes that the functional design of the reactivity control system conforms to the requirements of GDC 4, 23, 25, 26, 27, 28, and 29 and 10 CFR 50.62(c)(3) (as it relates to the alternate rod injection system and redundant scram air header exhaust valves scram capabilities) with regard to demonstrating the ability to reliably control reactivity changes under normal operation, AOOs, infrequent events, and accident conditions including single failures. The design of the reactivity control system conforms to the applicable acceptance criteria of SRP Section 4.6 and therefore is acceptable.

4.A <u>ESBWR Stability</u>

The staff focused its review of DCD Tier 2, Section 4D on whether the ESBWR design met regulatory requirements. The staff reviewed the applicant's methodology for calculating stability margins during the pre-application phase. In the SER within NEDC-33083P-A, the staff accepted the TRACG4 code and the applicant's associated methodology for calculating ESBWR stability margins.

Since the staff had previously reviewed the method for determining stability margins in detail, this evaluation focuses on a review of the ESBWR as it relates to meeting regulatory criteria for stability and stability during ATWS. Section 4A.1 below documents the staff's review of ESBWR stability and the basis for meeting regulatory criteria. Section 4A.2 documents the staff's review of ESBWR stability during an ATWS event and the basis for meeting regulatory criteria.

4.A.1 ESBWR Stability

4.A.1.1 *Regulatory Criteria*

The staff reviewed ESBWR stability (ESBWR, DCD Tier 2, Revision 9, Section 4D) based on the guidance in SRP Section 15.9, which lists the following high-level requirements for BWR stability reviews:

- GDC 10, as it relates to the reactor (reactor core, reactor coolant system, control and protection systems) being designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOO

- GDC 12, as it relates to power oscillations that can result in conditions exceeding SAFDLs are not possible, or can be reliably and readily detected and suppressed

- GDC 13, as it relates to a control and monitoring system to monitor variables and systems to assure adequate safety including those that can affect the fission process over their anticipated ranges for normal operation, AOOs, and accident conditions

- GDC 20, as it relates to, a protection system that automatically initiates the operation of the appropriate systems including the reactivity control systems, to assure that fuel design limits are not exceeded as a result of AOOs

- GDC 29, as it relates to the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs

- GL 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," dated July 11, 1994, as it relates to the need for reactors to install a stability long term solution (LTS) to satisfy GDC 10 and 12

Specific criteria applicable to the ESBWR design are described below.

To meet the requirements of GDC 12, the reactor core and its systems should be designed with sufficient margin to be free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of steady-state operation (including single-loop operation and extended cycle operation with reduced feedwater temperature where these operating conditions are proposed) and for AOOs. The design should consider the following:

- If potential oscillations cannot be eliminated, design proposals must detect and suppress (D&S) them reliably and readily.
- A reactor is considered stable if it satisfies one of the following criteria:
  - A. The calculated decay ratio (DR) for all three common stability modes (core wide, regional, and channel) satisfies the relationship DR less than (1 - σ) where σ is the uncertainty of the calculation. Staff must review and approve both the calculation methodology and its uncertainty. The value of σ is typically 0.2 but is methodology dependent. This value includes the code uncertainty and some degree of variability of the input parameters.
  - B. Use of an approved correlation to estimate the regional stability mode based on calculated core wide and channel DRs is permitted. One example is the stability criteria reviewed and approved by the staff and documented in NEDO-31960 "Long-Tem Stability Solutions Licensing Methodology," issued May 1991.

- The staff has reviewed and approved a number of stability LTSs. As reactor and fuel designs evolve, the industry may propose new stability LTSs. The following criteria judge the acceptability of new stability LTSs and facilitate meeting the requirements of GDC 20:

  - A. The LTS must protect against SAFDL violations automatically.
  - B. The LTS must demonstrate by analysis that either (i) the probability of instabilities in the allowed operating region is sufficiently small or (ii) unstable power oscillations can be detected and suppressed readily without SAFDL violations. The LTS may use a combination of both demonstrations for different instability modes.
  - C. If the licensing-basis option is declared inoperable, the LTS must provide a backup option, which may implement manual or administrative actions only if operator actions required to prevent SAFDLs can be accomplished within the 2 minutes allowed for operator action in the demonstration calculations.

D. The LTS option must include generic TSs that address:

   (i) The methodology for setpoint and region calculation and documentation of the setpoint on a cycle-specific basis (e.g., COLR)

   (ii) Operability and surveillance requirements for the licensing-basis option

   (iii) A time limit (120 days maximum) for operation under the backup option

- To meet requirements of GDC 13, stability-related instrumentation functionality must be demonstrated by analysis. Hardware implementation must follow SRP Section 7.2 requirements.

- In addition to the density wave instability modes, the applicant must ensure that the plant is free from other instability modes that could violate SAFDLs (e.g., startup or control system instabilities) or that oscillations can be detected and suppressed readily. Some instability modes may be acceptable with no potential for SAFDL violation (e.g., bi-stable flow or small flow oscillations during low-pressure startup).

4.A.1.2 *Summary of Technical Information*

To meet GDC 12, applicant used a stability criterion of DR less than 0.8 for all three density wave stability modes: core wide, channel, and regional. The applicant's criteria provide a DR margin of 0.2 to the ultimate criteria of a DR less than 1.0 to account for uncertainties. In addition, the applicant calculated the uncertainties in its best-estimate DR calculation using the code, scaling, applicability, and uncertainty (CSAU) methodology. The applicant applied this uncertainty to the DR calculated for normal operation with a feedwater temperature of 216 degrees C (420 degrees F) (point SP0 on SER Figure 15.1-3), effectively accounting for uncertainties twice.

ESBWR-specific analyses for the first core load demonstrate that unstable power oscillations are highly unlikely, thus complying with GDC 10. These calculations will be performed on cycle-specific bases as part of the reload analysis procedures to confirm the stability of the ESBWR for future cycles. As backup protection, the ESBWR design implements a defense-in-depth D&S solution based on the approved detect and suppress solution–confirmation density (DSS-CD) documented in NEDC-33075P-A, "General Electric Boiling Water Reactor Detect and Suppress Solution—Confirmation Density."

4.A.1.2.1 Density Wave Stability Results

In ESBWR DCD Tier 2, Revision 9, Section 4.D, NEDO-33337, and NEDO-33338, applicant presented the results of its stability analyses. To prevent density wave instabilities, applicant designed the ESBWR to have a very low DR during AOOs. DCD Tier 2, Revision 9, Tables 4D-2 through 4D-4 present the DRs for channel, super-bundle (16 fuel bundles), core, and regional oscillations for rated feedwater temperature operating conditions in an equilibrium core. NEDO-33337 and NEDO-33338 document the stability performance for the initial core at both rated and off-rated feedwater temperature conditions. This analysis demonstrated that the most limiting stability condition corresponds to the reduced feedwater temperature point (SP1M). The minimum allowable feedwater temperature for point SP1M (on SER Figure 15.1-3) is confirmed on cycle-specific bases to ensure that the calculated DR is less than 0.8 following a loss of feedwater heating anticipated occurrence from point SP1M.

4.A.1.2.2 Nondensity Wave Instabilities

The applicant identified two potential nondensity wave mechanisms for flow oscillations at low pressure (i.e., during startup). The first is a “geysering” flow oscillation, which results from vapor flashing at the top of the chimney region because the saturation temperature is lower at the chimney top than at the core because of the pressure difference. As the vapor flashing starts, core flow is increased and the core exit enthalpy is reduced, which stops the vapor generation, and a flow oscillation may occur. The other nondensity wave flow oscillation is the “Type 1” instability. These oscillations occur when there is voiding in the chimney, which leads to a reduction in the hydrostatic head in the chimney and an increase in flow. Oscillations of this kind are unavoidable in a natural circulation reactor because this instability region must be crossed before a steady two-phase voided region is established in the chimney. Applicant stated that the magnitude of these oscillations is small and the margin to critical power is very large; thus, these oscillations have no potential to violate SAFDLs and are acceptable under GDC 12.

In response to staff questions, applicant also evaluated the loop-type instability during normal operations by perturbing the chimney void fraction. Applicant showed that flow oscillations develop between the chimney and the downcomer, but they are highly damped, indicating that the ESBWR is not susceptible to oscillations from this mode that could potentially exceed SAFDLs, and therefore the oscillations are acceptable. For these calculations, a fine chimney nodalization scheme was used to minimize numerical damping.

4.A.1.2.3 Startup

In DCD Tier 2, Revision 9, Section 4D.2, applicant summarized a typical startup procedure and a TRACG analysis of the startup trajectory. Applicant presented the startup trajectory using an imposed core power (i.e., no neutronic feedback) with three different heatup rates. The lowest power level of 50 MW corresponds to a heatup of 30 degrees C/hour (54 degrees F/hour). The applicant stated that this is likely to be close to the actual value for startup. The median power level was 85 MW, with a corresponding heatup rate of 55 degrees C/hour (99 degrees F/hour). This is the highest allowable heatup rate so as to not exceed the reactor vessel thermal stress requirements. The highest power level the applicant used was 125 MW, which corresponds to a heatup rate of 82 degrees C/hour (147.6 degrees F/hour), which is above the allowable limit. Applicant showed large thermal margins to SAFDLs for the three heatup rates. Small-amplitude oscillations developed when voiding started at the top of the chimney; however, the core was still subcooled at that time and exhibited a large margin to CPR. Therefore, these oscillations do not have the potential of violating SAFDLs and satisfy the requirement of GDC 12.

4.A.1.2.4 Technical Specifications

TSs related to stability are part of the oscillation power range monitor, which implements the defense-in-depth solution as described in DCD Tier 2, Revision 9, Section 4D.3. The setpoints are cycle independent and are documented in DCD Tier 2, Revision 9, Table 4D.5.

4.A.1.2.5 Analysis Methodologies

The analysis method APPLICANT uses the TRACG coupled thermal-hydraulics three-dimensional neutronics code to analyze stability margins. TRACG is a time-dependent code with a full two-fluid representation. NEDE-33083P, Supplement 1, documents the TRACG04 code and APPLICANT analysis methodology for calculating stability margins in the ESBWR, and the corresponding SER presents the staff's approval. The stability analysis statistically accounts for the uncertainties and biases in the models and plant parameters using a Monte Carlo method for the normal distribution one-sided upper tolerance limit if the output distribution is normal, or the order statistics method if it is not. The application of the CSAU uncertainty methodology as it applies to stability is described in more detail in the response to RAI 4.3-22 and RAI 15.2-23. Based on the applicant's responses, RAI 4.3-22 and RAI 15.2-23 are resolved.

4.A.1.3 *Staff Evaluation*

The following sections document the staff's evaluation of the information presented by Applicant in ESBWR DCD Tier 2, Revision 9, Section 4.D, for the equilibrium core at rated feedwater temperature, and NEDO-33337 and NEDO-33338 for the initial core at both rated and off-rated feedwater temperatures. The staff followed the review procedures in SRP Section 15.9.

4.A.1.3.1 Applicability of the ESBWR Stability Criteria

Traditional BWRs (BWR/2-6) use a stability acceptance criterion on a two-dimensional map where core and channel DRs are set at limits of 0.8 and there is a cutout of the upper right corner of the defined rectangle where regional oscillations are expected to occur. This is sometimes referred to as the "dog-bite" correlation or the FABLE (a frequency domain stability code) criterion. When this criterion was established, no code was able to calculate the regional DR directly. Since TRACG is capable of predicting the regional DR, the staff requested that applicant calculate the ratio directly. Applicant implemented this change in response to RAI 4.4-10. In addition, applicant performed a Monte Carlo analysis of channel, core wide, and regional stability at rated power and flow and the limiting exposure for each stability mode. The limiting exposure is determined through iterative calculations found in Section 8.3.1 of NEDE-33083P-A, Supplement 1. Based on these calculations, the DCD reports the one-sided upper tolerance limit with 95 percent probability and 95-percent confidence level, which is roughly equivalent to a 2σ statistical treatment for normal distributions. From these calculations, the staff observes that the estimated TRACG uncertainty (at the 95/95 or 2σ level) in the DR is less than 0.2. The 0.8 DR acceptance criterion allows for 0.2 in uncertainties, and the applicant has demonstrated that this allowance is adequate. The acceptance criterion is conservative as both the predicted DR and the acceptance criterion itself include the uncertainties. Based on the applicant's response, RAI 4.4-10 is resolved.

4.A.1.3.2 Density Wave Stability Results

DCD Tier 2, Revision 9, Section 4D.1.3 presents the stability results calculated by TRACG for the candidate ESBWR plant design, with 1,132 bundles and a rated thermal power of 4,500 megawatts thermal, operating at rated feedwater temperature. The TRACG ESBWR model includes 24 thermal-hydraulic regions plus 4 hot channels. The TRACG core-wide model uses a different channel grouping but the regional mode results tend to be the limiting case for ESBWR stability evaluations.

GEH conducted analyses of various points of an equilibrium GE14E cycle: BOC, MOC at the peak hot excess (PHE) reactivity point, and EOC. The predicted DRs under steady-state conditions for ESBWR using TRACG are well within the acceptance criteria (DR less than 0.8). The DRs calculated by staff confirmatory LAPUR calculations are similar, and range from 0.12 to 0.24. These DRs are very small (very stable conditions) and hard to estimate accurately. One-to-one comparisons between calculations are not possible because the DR "estimation" error dwarfs all other errors at these low values. The conclusions of this review are based on the fact that both TRACG and LAPUR predict similarly low DRs at the rated feed water temperature.

Applicant also conducted analyses for an initial core at off-rated feedwater conditions. The analyses indicate that low feedwater temperatures result in lower margin to stability. This is caused by a shift of the axial power shape to the bottom of the core.

Indeed, as described in NEDO-33338, stability considerations limit the minimum feedwater temperature allowed for operation. The feedwater temperature of point SP1M (on SER Figure 15.1-3) is defined so that the DRs calculated following a loss of feedwater heater transient are less than the 0.8 criteria. This calculation is to be performed on a cycle-specific basis and the minimum allowed feedwater temperature is to be reported in the COLR.

Stability is a crucial design requirement for the ESBWR because the rated power and flow conditions are the limiting conditions for stability during normal operation. However, following an AOO, the power/flow conditions could be even more severe than at rated conditions. Therefore, AOO analyses must be reported in the COLR an evaluation of stability. In general, the stability margin reduces when the reactor power increases and/or core flow reduces. Because the ESBWR design relies on natural circulation for core flow circulation, the core flow during full-power operation depends only on the vessel water level. Higher water level means higher core flow and vice versa. During normal operation, the water level is tightly controlled, and a reactor scram is initiated when the water level is too high or too low.

DCD Tier 2, Revision 9, Section 4D.1.5 identifies two AOOs with the potential to decrease the ESBWR stability margin: LOFWH, which results in increased power; and loss of feedwater flow (LOFW), which results in a lower flow. DCD Tier 2, Revision 9, Table 4D-4 shows the DRs calculated by TRACG for these events when the ESBWR is operating at rated feedwater temperature; the most limiting event is the increase in power caused by the LOFWH. The core DR increases by about 0.14, but it remains well below the acceptance criteria. The LOFW is a milder event because the scram system trips the reactor when the water level reaches the Level 3 setpoint. Operation with reduced feedwater temperature results in a decrease of stability margin, and the LOFWH from point SP1M is the limiting stability event, with a calculated regional DR of 0.71 (see Table A.1-3 of NEDO-33338). This calculation assumes a reduction of 16.7 degrees C (30 degrees F) in feedwater temperature; larger temperature reductions would result in SCRRI initiation and suppression of the event.

The staff review concurs with the applicant’s evaluation of the effects of AOOs on ESBWR stability margins. The results meet the acceptance criteria discussed in Section 4.A.1.1 of this report, and the calculations show that the ESBWR is stable under the postulated AOOs.

In RAI 4.4-57, the staff requested that the applicant provide regional mode DRs for the two limiting AOOs. In response, the applicant provided an evaluation of regional DRs during AOOs. The applicant also updated DCD Tier 2, Table 4D-4 on the basis of these results. The analyses indicate that the regional DR is limiting for AOOs. Based on the applicant’s response, RAI 4.4-57 is resolved.

4.A.1.3.3 Nondensity Wave Instabilities

The staff reviewed the potential for nondensity wave instabilities in the ESBWR. Considering the startup instabilities identified by the applicant (geysering and Type 1), the staff agreed that

Type 1 instabilities will occur during startup. However, these will not pose a challenge to SAFDLs because of the large margins and low power during startup, and therefore the staff finds that these instabilities are acceptable during startup and are not inconsistent with GDC 12. Section 4A.1.3.4 of this report discusses the startup.

The staff also considered the potential for loop-type (or buoyancy-driven) oscillations during normal operations. The staff requested that the applicant perturb the buoyancy term in the chimney to confirm that chimney oscillations do not develop. A fine nodalization scheme was used for these calculations to avoid numerical damping. The oscillations damped immediately. In RAI 4.4-58 S01, the staff requested the applicant to explain the apparent differences between TRACG04 results and experimental results in the GENESIS facility. The applicant's response indicated that TRACG04 reproduces the GENESIS experimental results when the neutronic feedback in TRACG04 is turned off, simulating the electrically heated bundles in the experimental facility. In response to RAI 4.4-58 S01 and RAI 4.4-11, the applicant provided data supporting the assertion that the ESBWR chimney has no significant effect on stability. These chimney results are independent of the chimney nodalization (coarse or fine). The results of these calculations show that loop oscillations driven by chimney buoyancy perturbations are not likely to develop in the ESBWR. Based on the applicant's responses, RAI 4.4-58 and RAI 4.4-11 are resolved.

If the ESBWR is operated close to a flow-regime transition boundary, it is conceivable that an oscillatory instability may develop. The staff considered the potential for flow-regime transition instabilities to develop in the ESBWR. At rated power, the ESBWR is expected to have fully developed churn-turbulent flow, except for possibly a few low-power periphery partitions. In addition, there will be thermal-hydraulic communication between all of the chimney partitions and channels via the core bypass, which will tend to equalize the partition void conditions. In response to RAI 4.4-39, applicant confirmed that the pressure at the outlet of the core will be uniform across the core. This is because the core outlet (and chimney inlet) conditions communicate hydraulically via the liquid level in the core bypass. The staff issued RAI 4.4-39 S01 requesting applicant to evaluate the bypass flow conditions. The staff concluded that flow regime transition oscillations will not be a concern in the ESBWR at rated conditions. The staff disagreed with the applicant's assertion that the TRACG and PANACEA calculations are independent, based on information provided in the response to RAI 21.6-85. In RAI 4.4-39 S02, the staff requested that the applicant perform an analysis to determine the core outlet pressure distribution using an independent verification approach. The applicant provided a TRACG calculation that uses an initialization process that is independent of PANACEA. The results of this calculation confirm that core outlet pressure is uniform. Based on the applicant's response, RAI 4.4-39 is resolved.

In RAI 14.2-89, the staff requested that, during startup testing, the COL holder characterize the power levels at which flow-regime transition oscillations may possibly occur. The staff recommended that the licensee analyze the neutron flux from LPRMs under each chimney partition. Also, the staff requested that applicant develop a startup testing plan to identify the impact, if any, of operation at reduced power levels where oscillations induced by flow transition may be possible.

In response, the applicant revised the DCD Revision 5, Section 14.2.8.2.7, identifying this test as a initial test program (ITP) in order to identify the impact of any possible flow oscillations. The applicant also committed to developing an additional single plant startup test based on LPRM readings. This ITP is also identified in the DCD Tier 1, Section 3.5, and represents a commitment that combined operating license applicants referencing the certified design will implement an ITP that meets the objectives presented above. Based on the applicant's response, RAI 14.2-89 is resolved.

In response to RAI 21.6-113, the applicant argued that chimney entrance effects and flow at the chimney inlet that is not fully developed could have two separate effects: (1) alteration of the steady-state void fraction or (2) induction of time-dependent fluctuations (i.e., noise) in the void fraction. The applicant evaluated the steady-state void effects and concluded that the real void fraction at the chimney inlet may be lower than that calculated by TRACG04. The applicant presented experimental evidence suggesting that the length of the entrance region is small (approximately one equivalent diameter) relative to the chimney height such that the effect on the calculated chimney static head and natural circulation flow is small. The experimental evidence includes (1) data from the Dodeward reactor and (2) data from Dubrovskii, which covers the reactor operating pressure (75 bar [1,088 psi]) and has a similar diameter (0.61 meters [2.0 ft]). Based on these data, the applicant concluded that the steady-state void fraction at the chimney inlet could be as low as 75 percent of the fully developed void, but the region not fully developed is at most 1 m (3.3 ft) long. The applicant performed a calculation assuming 70 percent of the fully developed void for 1 m (3.3 ft) and determined that the impact on recirculation (core) flow is less than 3 percent. This 3-percent overprediction is an upper-bound estimate, and the actual flow error is expected to be smaller.

The staff concurred with the applicant's evaluation. The available data indicate that entrance effects and not fully developed flow may reduce the steady-state void fraction by up to 25 percent, but only for the first meter of chimney. The effect on recirculation core flow of this misprediction is small (less than 3 percent) and should not have any significant effect on TRACG calculations.

The applicant presented experimental evidence of the impact of time-dependent void fraction fluctuations (i.e., noise), especially in the churn-turbulent regime. The data presented include an evaluation of the Dodeward reactor data and the Dubrovskii data. The applicant reported that no significant flow oscillations were observed in the experimental data. Thus, the applicant concluded that void fraction oscillations caused by turbulence in the churn flow regime will have little or no effect because if they are fast, they will be averaged out in the chimney. If they are slow, they will be compensated for by changes in core exit void to maintain the reactor critical. In addition, both the Dodeward and Dubrovskii experimental data indicate negligible flow oscillations. Both sets of experimental data are in the churn-turbulent flow regime. The staff concurs with the applicant's evaluation. Based on the applicant's response, RAI 21.6-113 is resolved.

The staff reviewed the information presented by the applicant related to flow oscillations in the ESBWR chimney. This information included (1) TRACG04 calculations with a detailed axial nodalization, (2) a TRACG04 benchmark against the GENESIS experiment, and (3) an evaluation of the chimney entrance effects and flow regime oscillations using experimental data. The staff concurs with the applicant's evaluation that loop oscillations driven by chimney buoyancy perturbations are not likely to develop in the ESBWR.

4.A.1.3.4 Startup

During normal operation, the stability mode of concern is the so-called density wave that produces flow and power oscillations within a frequency range between 0.5 and 1 hertz (Hz). Because of its unique startup process, other instability modes are of concern during ESBWR startup. These instability modes include geysering instability and loop instabilities (also known as manometer or Type I instabilities). The TRACG capability of modeling both of these modes was reviewed and accepted in NEDE-33083P-A, Supplement 1.

The key in the startup procedure is maintaining power low enough so that boiling occurs only at the top of the chimney and not inside the active core. By maintaining voids out of the core at low pressure, the ESBWR prevents reactivity feedback issues, which could result in violent power oscillations.

As the circulating water is slowly heated, saturation temperature is first reached at the top of the separators because the pressure is lower, given the density head or weight of the column of water in the chimney. Vapor generation at the top of the separators results in a reduction in the chimney density, which reduces the pressure causing the voiding front to propagate downward. The formation of voids also results in a larger driving head for natural circulation flow. The increase in natural circulation flow reduces the core exit temperature and leads to a collapse of the voids. This completes one cycle of the hydrostatic head oscillation, and these oscillations persist until the temperature of the water inventory in the core increases and a steady void fraction is established in the separators. Small oscillations in the flow rate are harmless when the power is low and the core flow is single phase, and consequently, thermal limits have a very large margin.

The applicant simulated the ESBWR startup procedure with TRACG and demonstrated that the ESBWR proposed startup procedure is feasible. The results showed no significant power oscillations even for heatup rates larger than allowed by TSs. CPR limits were not violated by any of these scenarios.

Oscillations do develop during the startup as Type 1 (manometer type) instability. These oscillations can be seen as a rapid variation of void fraction in the separators. Because the core coolant is subcooled at the time of the oscillations, the margin to boiling transition is very large. Flow oscillations in subcooled regimes are of no consequence to the SAFDLs. GDC 12 specifies that “The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.” Since the predicted Type 1 ESBWR instabilities have no potential to exceed SAFDLs, the staff concludes that their presence is not inconsistent with GDC 12, and these oscillations are acceptable.

In RAI 4.4-59, the staff requested that applicant establish a maximum heat-up rate for the low pressure start-up in terms of MW per hour that will not be exceeded by a Licensee. in addition, the staff asked GEH to show adequate margin to instability by simulating the start-up path using a larger heat-up rate that bounds the established maximum. The applicant was asked to use neutronic feedback.

In response, the applicant performed a detailed analysis of the ESBWR startup with a wide range of parameter variations to attempt to bound the expected startup conditions. For all these simulations, neutronic feedback was simulated, as requested. The study concluded that startup rates as high as 110 degrees C/hour (198 degrees F/hour) are safe and free from instabilities that could challenge SAFDLs. This demonstrated a safe startup value that is twice as large as the maximum heatup rate allowed by the thermal-stress limit of 55 degrees C/hour (99 degrees F/hour), and approximately four times larger than the expected ESBWR heatup rate of 27.5 degrees C/hour (49.5 degrees F/hour). Based on the applicant’s response, RAI 4.4-59 is resolved.

In the SER for NEDE-33083P, Supplement 1, the staff noted that for the ascension to full-power phase of startup, which occurs approximately 8 hours into the startup, the current approach for modeling stability analyses does not include a balance-of-plant model. As documented in the SER for NEDE-33083P, Supplement 1, the feedback from steam flow into the feedwater system would be necessary in order to perform best-estimate analyses of the transient response to an oscillation over long periods. In addition, xenon (Xe) would have a more pronounced effect on the power distribution. In RAI 4.4-60, the staff requested that the applicant provide a calculation demonstrating margins using transient Xe and include a balance-of-plant model for the startup through ascension to full power.

In response, the applicant did not perform the requested TRACG calculation. Instead, a series of "PIRT46" TRACG calculations were used to simulate the Xe effect via the impact on local power peaking. The applicant presented a series of physical arguments to justify why the constant-Xe calculation is acceptable. The basis for these arguments is that a typical startup from cold shutdown to full pressure takes about 5 hours, and Xe burnup is not significant at less than 2-percent power for 5 hours.

In the response, the applicant stated that TRACG does not calculate time-varying Xe. It uses a constant cross-section set generated by PANAC11 for a given Xe condition. TRACG does not calculate time-varying Xe, but in the "PIRT46" parameter, it provides a capability to simulate Xe effects by increasing or decreasing local power peaking. The applicant performed a PANACEA study for the ESBWR initial core at MOC. PANACEA, being a series of steady-state calculations for the startup path, can model the Xe burnup. Based on these PANACEA calculations, the applicant concluded that a radial peaking factor (RPF) of 8 conservatively bounded the expected radial peaking when Xe burnup is accounted for (the nominal RPF value is approximately 5). For the TRACG calculations, RPF values as high as 11 were used. In the nominal case, an RPF value of 5 was used, which corresponds to a hot channel power of 479 kilowatts (kW) (for a heating rate of 90 MW for the core). In the Xe burnup bounding simulation, the RPF was increased to 11 (hot channel power 1,440 kW) and the resulting minimum CPR was reduced from 7.2 to 5.3. A CPR margin of 5.3 is a very significant margin. Thus, the staff concludes that Xe burnup effects are not likely to invalidate the conclusion that SAFDLs will not be exceeded during startup. Thus, the GDC 12 requirements are satisfied even when Xe burnup is accounted for. Based on the applicant's response, RAI 4.4-60 is resolved.

4.A.1.3.5 Effect of Chimney Models

In response to staff RAIs, applicant performed a series of detailed analyses of the effect of the chimney on the density wave and loop stability modes. The ESBWR TRACG model was modified to include a fine node structure in the chimney region. The analyses included a case with a core wide power response to a pressure perturbation and cases with buoyancy perturbations. The staff concludes that the finely nodalized chimney allows for a more accurate representation of void propagation through the chimney but has no effect on the stability results. Even though applicant stated that the original nodalization used for the stability calculations in the DCD are adequate for stability analyses, the staff recommended in RAI 4.4-58 that the TRACG model with the fine chimney nodalization be used for future ESBWR stability calculations. Applicant responded to this RAI by stating, "In summary, the finely nodalized chimney allows for a more accurate representation of void propagation through the chimney, but has no effect on the stability results. The original nodalization used for the stability calculations in Reference 4.4-11-1 and the DCD is adequate for stability analysis." The calculation discussed in the response to RAI 4.4-58 applies to the ESBWR and shows that results are insensitive to the nodalization model. Therefore, applicant does not believe it is necessary to perform stability calculations in support of the DCD with the fine nodalization chimney model of TRACG to guarantee that chimney oscillations do not affect the core stability.

The staff issued RAI 4.4-58 S01, which pointed out an apparent incompatibility of results between TRACG calculation and experimental data from the GENESIS facility. GENESIS is a thermal-hydraulic loop simulation of the ESBWR with a single channel and a long chimney. The power to the channel may be modulated by a computer-simulation of the reactivity feedback based on online void fraction measurements. In the GENESIS facility experiments, a low frequency of oscillation (approximately 0.1 Hz) was observed when the power to the channel was maintained constant. This is an approximation of the purely thermal hydraulic or "channel" oscillation mode simulated by TRACG. The TRACG results did not agree with the experimental data and showed a significantly larger oscillation frequency (approximately 0.8 Hz). These results indicated that the chimney did not take part in the TRACG oscillation, while the GENESIS results indicated that the chimney does take part in the oscillations because of the lower oscillation frequency.

In response to RAI 4.4-58 S01, the applicant performed a TRACG04 simulation where the chimney buoyancy term was perturbed. An oscillation of about 0.1 Hz was observed when the channel power was maintained constant, simulating the GENESIS results. The applicant concluded that there is no discrepancy between the GENESIS and TRACG04 results and that TRACG04 can model loop-type oscillations in the chimney.

In the second part of the response to RAI 4.4-58 S01, the applicant justified the use of coarse nodalization in the chimney. The applicant argued that the chimney does not play an important role in the density wave instabilities of interest. Loop oscillations (where the chimney plays an important role) are not limiting in the ESBWR and do not pose any significant safety concern. The applicant concluded that the coarse chimney nodalization was adequate for ESBWR stability analysis.

After review of the available data, the staff finds that (1) when using fine nodalization, TRACG can model the loop-type buoyancy-driven flow oscillations that were observed in the GENESIS experiment, (2) both TRACG04 and GENESIS are in relatively good agreement in predicting the frequency and DR of chimney loop-type oscillations, and (3) for the density wave oscillations that are likely to be limiting in the ESBWR, the chimney does not appear to play a significant dynamic role, and thus, numerical damping in the chimney region is not likely to affect the magnitude of the calculated DR.

Therefore, the staff concurs with the applicant's evaluation and accepts that a coarse chimney nodalization would be sufficient to model density wave oscillations. Based on the applicant's response, RAI 4.4-58 is resolved.

4.A.1.3.6 Stability Long-Term Solution

DCD Tier 2, Revision 3, Section 4.3.3.6.2 stated that a D&S solution is the preferred option for the ESBWR.

In RAI 4.3-7, the staff requested applicant to provide a detailed description of the stability solution chosen for the ESBWR, whether it needs further staff review or it is a standard solution, associated TSs, and how the TSs reflect the setpoint calculation (if any). The applicant responded that it selected the standard D&S solution DSS-CD, as documented in the NRC-approved GEH proprietary report NEDC-33075P, Revision 5. The applicant also provided a proposed DCD revision to incorporate DSS-CD into the ESBWR TS Sections 3.3.1.4 and 3.3.1.5 with cycle-specific setpoints for the DSS-CD system to be provided in the individual plant COLR, as specified in TS 5.6.3. In addition, the applicant updated the DCD in Revision 5 to include Section 4D-3, which describes the ESBWR-specific features of the defense-in-depth D&S solution proposed for the ESBWR. This solution uses all the approved algorithms from DSS-CD, with parameter settings adjusted to the special ESBWR characteristics. Because it is a defense-in-depth measure, a licensing-basis calculation is not required to demonstrate the effectiveness of the solution in preventing SAFDLs. The staff concluded that the ESBWR defense-in-depth solution satisfies the requirements of GDC 13, 20, 29 and GL 94-02 because the ESBWR defense-in-depth is implemented in the already approved DSS-CD and Option III hardware, which satisfies the instrumentation and controls and hardware requirements of GCD 13.

The ESBWR defense-in-depth D&S solution is a defense-in-depth feature, but in the case oscillations were to develop, it would initiate an automatic scram, which satisfies the requirements of GDC 20 and 29.

The licensing basis of the ESBWR is demonstration of stability by analysis; therefore, the ESBWR implements a Solution I type of LTS. Through TRACG04 analyses, ESBWR operators will demonstrate on a cycle-specific basis that the ESBWR will always operate outside of the stability exclusion region. In addition, the defense-in-depth solution will provide a D&S (Solution III type) feature as defense-in-depth. A backup stability solution is also provided as required by GL 94-02. Thus, the staff concludes that the ESBWR stability methodology satisfies the requirements of GL 94-02. Based on the applicant's response, RAI 4.3-7 is resolved.

In RAI 4.3-8, the staff pointed out that all approved D&S solutions have an armed region. Typically, the solution is only armed for low-flow maneuvers and represents a small fraction of the cycle time. Since the ESBWR operates at the equivalent of low-flow at nominal conditions, one expects that the D&S solution must remain armed for the complete cycle. The staff requested applicant to discuss the armed region implications and the associated probability of false alarms.

In response to RAI 4.3-8, the applicant described the stability LTS armed region. The armed region will include normal operation, and it will be defined in the COLR based on power and feedwater temperature. With regard to the probability of false alarms, the applicant stated in the response, that ESBWR is free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of normal operation. Therefore, the D&S solution is implemented to provide Defense-In-Depth protection from instabilities that are not anticipated. Avoidance of spurious scram, therefore, is achieved by elevating the threshold set point of the oscillation detection algorithm of D&S well above the level of measured LPRM background noise. Therefore, based on the applicant's response, RAI 4.3-8 is resolved.

In RAI 4.3-9, the staff indicated that a future licensee may have the flexibility to deviate from the standard certification and choose a different long-term stability solution. The staff requested the applicant to specify criteria that must be met by the ESBWR for long-term stability solutions.

In response, as referenced in DCD Tier 2, Chapter 4, Section 4D and Chapter 16, Specification 3.3.1.4, the applicant stated the requirements that a future LTS must meet. These commitments are that (1) GDC 12 will be satisfied, (2) the LTS will provide a backup solution in case the primary solution is declared inoperable, and (3) the backup solution will not be active for longer than 120 days. Based on the applicant's response, RAI 4.3-9 is resolved.

4.A.1.3.7 Analysis Methodologies

The applicant used the TRACG04 code and methodology as documented in NEDE-33083P, Supplement 1, to calculate stability margins for the ESBWR. The staff reviewed this methodology during the pre-application phase of the ESBWR and accepted it with open items. The staff review of the open items for this method is documented as an addendum to the SER for NEDC-33083P-A, and is also summarized in Section 21.6 of this report.

For analysis in support of the DCD, the applicant used three different versions of the TRACG04 code to generate stability results. The primary reason for the use of the different code versions was the date of the calculation. All code versions were "non-level 2" and were used only after a validation of the code against experimental data previously performed. These "external-data validations" served the purpose of alternate calculations as required by the applicant's Engineering Operation Procedure. The applicant used the following TRACG04 versions:

- T4N2, which corresponds to version 45 in the Alpha platform, this version was used for the early DCD calculations. Three exposures were calculated (BOC, MOC, and EOC). In addition, the original "stability during startup" calculation was performed with this version assuming a constant power generation (no neutronic feedback).

- T4N3, which corresponds to version 49 in the Alpha platform, this version was used to respond to staff RAIs related to the earlier T4N2 calculations. The stability during startup calculation was also updated including three-dimensional neutronic feedback, as requested by the staff. Version 49 was used for this calculation because it was the most recent, validated version of TRACG at the time.

- T4PN53, which corresponds to version 53 in the PC platform, this version was used to respond to staff RAIs related to the earlier T4N2 calculations. In response to these RAIs, the applicant performed a number of stability calculations around the MOC point with a fine mesh of exposures to identify the maximum DR as function of exposure. The PC V53 version was used because (1) it was the most recent version and (2) the PC version is significantly faster and allowed to perform the many calculations required to step through the exposure fine mesh to identify the maximum DR.

4.A.1.3.8 Staff's Independent Calculations

The staff performed independent calculations using the LAPUR code to evaluate the stability of the ESBWR. LAPUR is a frequency domain code developed by Oak Ridge National Laboratory that is used for BWR stability analysis. The staff performed calculations at 12 points of a representative fuel cycle at nominal feedwater temperature using the design information in the DCD. The LAPUR confirmatory calculations showed that the ESBWR stability is within the limits of the design criteria.

The highest calculated DR at nominal feedwater temperature is 0.24, and corresponds to the core wide stability mode for the EOC condition, when the axial power shape becomes flat or slightly top peaked. The LAPUR results were in good agreement with the ODYSY and TRACG results reported by the applicant for the nominal operating conditions. The LAPUR confirmatory calculations also indicated that the dynamic model used to simulate the chimney riser has little or no effect on the stability of the ESBWR. The riser itself has a large effect on the core flow, but it has a very small friction pressure drop. However, once the core flow and power are fixed, the presence of the chimney does not influence stability. Accordingly, the chimney plays a crucial role in setting up the steady-state value of the core flow, but plays only a minor role during the unstable oscillations. As a result of its calculations, the staff concludes that the ESBWR DR is within the limits of the acceptance criteria.

4.A.2 ESBWR Stability during Anticipated Transient without Scram

Chapter 15 of this report contains the major part of the review of the ATWS event scenario. This section addresses the issue of thermal-hydraulic stability during an ATWS scenario.

4.A.2.1 *Regulatory Criteria*

The staff based its review of ESBWR stability performance during an ATWS event on SRP Section 15.8, which lists the following procedure to be used for BWR ATWS/stability reviews:

For BWRs, the ATWS/stability evaluation was addressed generically in topical reports NEDO-32047 and NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," which defines the ATWS mitigation actions for plants operating up to original licensed thermal power. SRP Section 15.8, III.6 gives the following guidance:

A. For all applications, the reviewer will evaluate the implementation of the ATWS/Stability Mitigation Actions in design-specific (Emergency Procedure Guidelines [EPGs]), or plant-specific EOPs or plant-specific Emergency Operating Instructions (EOIs). The reviewer will ensure that sufficient information has been provided to justify that the mitigation actions are effective in maintaining core coolability criteria for the limiting ATWS/Stability event.

B. For BWRs that implement extended power uprate (EPU) and expanded power-flow domains (e.g., Maximum Extended Load Line Limit Analysis (MELLLA+)), the licensee will demonstrate that the ATWS/Stability Mitigation Actions are effective in maintaining core coolability criteria for the limiting ATWS/Stability event.

C. For evolutionary BWRs, the licensee will provide EOPs or EOIs that implement ATWS/Stability Mitigation Actions equivalent to those approved in Reference 8 [NEDO-32047 and NEDO-32164], including manual boron injection if oscillations are detected. The licensee will demonstrate the EOPs or EOIs are effective in maintaining core coolability criteria for the limiting ATWS/Stability event.

4.A.2.2 *Summary of Technical Information*

To demonstrate that there are no stability issues during an ATWS transient for the ESBWR, applicant stated that the ATWS mitigation features for the ESBWR include automatic feedwater runback and automatic boron injection. The applicant simulated an ATWS event for the MSIV closure event using TRACG04. In response to RAI 21.6-45, the applicant described the method used to perform this calculation. During this event, the applicant introduced a flow perturbation at the inlet of the channels during the transient and showed that the ATWS acceptance criteria are satisfied even though a small-amplitude power oscillation was observed. Based on the applicant's response, RAI 21.6-45 is resolved.

4.A.2.3 *Staff Evaluation*

To demonstrate acceptable performance, a limiting ATWS scenario must be investigated. In this case, the limiting ATWS scenario is that which leads to the greatest magnitude oscillation and it is treated as a separate requirement from those dictated by SRP Chapter 15. The staff agrees with the applicant in the selection of the MSIV closure ATWS event as the limiting event for the Chapter 15 analysis, as this particular event simultaneously challenges the system integrity with high neutron flux, high vessel pressure, and high suppression pool temperature.

However, the staff had previously reviewed TRACG calculations of ATWS instability events for operating BWRs and determined that conditions exist for particular ATWS scenarios where instability events are, in fact, likely. These scenarios are those that result in high power and low flow. An isolation event, such as MSIV closure, will result in a rapid increase in reactor pressure, which leads to the actuation of the DPVs. The ensuing depressurization reduces reactor power. For a conventional BWR, because of the SRVs, instabilities are more likely to occur when the system is not isolated.

Again for conventional BWRs, when recirculation pumps trip (reducing flow) or on a loss of feedwater heat (increasing power and shifting power towards core bottom), the system becomes more susceptible to thermal-hydraulic instability as there is a high power-to-flow condition following in either of these events. The downward shift in axial power following an LOFWH reduces the single-phase to two-phase pressure drop ratio, thereby further reducing the stability margin. Similarly, turbine trip with full bypass may produce a pressure perturbation that will impact core reactivity by collapsing voids at the initiation of the transient, yet not initiate an isolation of the RPV. Therefore, the analysis of ATWS stability should be addressed using a limiting transient from the perspective of core stability.

The ESBWR ATWS mitigation actions include the following:

(1) A reliable RPS with two redundant methods of inserting control rods: (a) hydraulic rod insertion and (b) electrical FMCRD insertion. By reducing common-cause failure mechanisms, these redundant systems make the probability of failure to scram small

(2) An alternate rod insertion function, which uses sensors and logic that are diverse and independent of the RPS, as required by the ATWS rule

(3) Automatic feedwater runback, which reduces the reactor water level and the core power generation. This function is a substitute for the recirculation pump trip required by the ATWS rule

(4) Automatic initiation of standby liquid control, as required by the ATWS rule for new reactors

The ESBWR hardware design described in actions (1) and (2) above reduces the probability of a failure to scram. Actions (3) and (4) are an implementation of the ATWS/stability mitigation actions in operating reactors. The ESBWR design automatically implements the EPG ATWS/stability mitigation actions without the need for operator intervention. With these mitigating actions, the ATWS/stability event will not be allowed to progress, and large-amplitude unstable power oscillations are not likely to develop in the ESBWR.

Because the ESBWR design does not include recirculation pumps, the staff requested that the applicant select the LOFW accident and turbine trip with full bypass as the events for predicting the system performance during an ATWS instability event. Each event is a nonisolation event resulting in increased reactor power. In the case of LOFWH, the increased reactor power comes from an increase in coolant subcooling and hence an increase in moderator density. In the case of a turbine trip with full bypass, a momentary pressure wave sent down the steamline leads to a momentary reduction in core void content. In RAI 21.6-51, the staff requested that applicant use the approved methodology in the SER for NEDE-33083P, Supplement 1, to perform a DR calculation or to add margin by increasing the void reactivity coefficient.

In response, the applicant concluded that the most limiting ATWS event from the point of view of stability is turbine trip with bypass (TTWB). The applicant argued that TTWB is more limiting than loss of feedwater (LOFW) because LOFW only reduces the water level, while TTWB reduces the water level (because of the feedwater runback) and increases the subcooling significantly. Nevertheless, in spite of this evaluation, the applicant presented results for both the TTWB and LOFW.

The ATWS/stability evaluation was performed using a regional-mode channel grouping scheme, which does not preclude core wide oscillations and is, therefore, more general. In addition, a 130-percent multiplier was added to the density reactivity coefficient to increase the conservatism. Figure A.4.2.1-1 of NEDO-33338, Revision 1, shows the result of this calculation. The TRACG04 ATWS analysis shows that, in all of these conservative calculations, the ESBWR is slightly unstable under ATWS conditions because a small-amplitude regional limit cycle is observed early in the transient. Approximately 85 seconds into the transient, the automatic water-level reduction results in uncovering of the steam separators, and the self-sustained limit cycle oscillation decays as the ESBWR becomes once again stable without operator intervention. The TRACG04 calculations indicate that CPR or other limits were not violated during this bounding ATWS transient.

The staff concludes that the ATWS criteria are satisfied even though a small-amplitude power oscillation was observed. Based on the applicant's response, RAI 21.6-51 is resolved.

In conclusion, the staff finds that large-amplitude unstable power oscillations (ATWS/stability) that could compromise ATWS criteria are not a likely event in the ESBWR because (1) the ATWS/stability mitigation actions are implemented automatically and (2) the low probability of a failure to scram.

4.A.3 Conclusions

A summary and the major conclusions from the staff's review are provided below:

(1) The stability criteria set forth in the DCD comply with the guidelines in SRP Section 15.9. The acceptance criteria for calculated DRs for the three density wave instability modes are the following:

   a. Channel DR less than 0.8

   b. Core wide DR less than 0.8

   c. Regional DR less than 0.8

(2) The ESBWR DRs will be calculated using TRACG and the methodology documented in NEDE-33083P-A, Supplement 1, or an alternate methodology which has been reviewed and approved by the staff for use in ESBWR applications.

(3) The ESBWR DR values used in the acceptance criteria for the rated feedwater temperature conditions include a one-sided upper tolerance limit with 95-percent content and 95-percent confidence level. The uncertainty values are determined by a Monte Carlo analysis using the CSAU methodology. This is an acceptable treatment of uncertainties, and it is conservative because the acceptance criteria already contain a 0.2 margin to account for variability in modeling assumptions.

(4) Applicant calculations and staff confirmatory calculations indicate that the ESBWR satisfies the stability criteria at rated feedwater temperature conditions. The largest estimated DR is 0.53 (regional mode, MOC).

(5) Applicant calculations and staff confirmatory calculations indicate that the ESBWR satisfies the stability criteria at off-rated feedwater temperature conditions. The largest estimated DR is 0.61 (at point SP1M, regional mode, MOC).

(6) The two limiting AOOs are (1) LOFWH, which increases power to the scram setpoint, and (2) LOFW, which reduces core flow until the low-water-level setpoint is reached. The highest calculated DR during AOOs that start at rated conditions is 0.66, and it corresponds to the LOFWH at MOC for the core wide instability mode. An LOFWH event initiated at off-rated conditions (point SP1M) results in a DR of 0.71. These values are within the acceptance criteria.

(7) The DCD presents an evaluation of the stability during an ATWS. The staff concurs with applicant's evaluation that stability during an ATWS is not a concern in the ESBWR for the following reasons:

   a. The immediate water-level reduction caused by the automatic feedwater runback reduces the power and flow rate, and it exposes the feedwater to vessel steam; therefore, the large subcooling transient that causes the ATWS/stability event in operating reactors does not occur in the ESBWR.

   b. The automatic boron injection and, most important, the direct injection into the core bypass area reduce the duration of the ESBWR ATWS so that unstable power oscillations will be highly unlikely.

(8) Two types of startup instabilities have been evaluated by applicant for the ESBWR: geysering and Type 1 (or manometer). The staff concurs with applicant's evaluation that these instabilities will occur during startup, but will not pose a challenge to SAFDLs. These types of instabilities are acceptable and are not inconsistent with GDC 12.

(9) In addition to the density wave and startup instability modes, applicant has evaluated the loop-type instability mode by perturbing the chimney void fraction at power. Flow oscillations develop between the chimney and the downcomer, but they are highly damped, showing that this oscillation mode is very stable.

(10) To evaluate the effect of the chimney, applicant has set up a TRACG model with fine nodalization in the chimney region. In this model, the Courant = 1 limit occurs in the

chimney nodes. Stability evaluations with this model and previous models with coarser nodes show no significant difference. The staff confirmatory calculation using the LAPUR code confirms these results. These calculations indicate that the chimney dynamics play a very minor role in density wave oscillations.

(11) The staff performed confirmatory calculations to determine the power level at which the chimney will transition from slug/churn to annular flow. Oscillations may occur at the flow regime transition power. The staff calculations indicated that the flow regime transition will occur between 30-percent and 70-percent power. Rated conditions will have fully developed annular flow (except for, possibly, a periphery channel). Thus, the staff concludes that flow regime transition oscillations will not be a concern in the ESBWR at rated conditions.

All stability-related open items are resolved. Based on the preceding review, the staff concludes that the plant design adequately addresses stability issues and satisfies all the criteria specified in SRP Section 15.9 and, specifically, GDC 10, 12, 13, 20, and 29; Appendix A to 10 CFR Part 50; and GL 94-02.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Introduction

The reactor coolant system (RCS) includes those systems and components that contain or transport fluids coming from or going to the reactor core. These systems form a major portion of the reactor coolant pressure boundary (RCPB). This chapter of the safety evaluation report (SER) describes the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the RCS and the pressure-containing appendages out to and including the outboard isolation valves. This grouping of components, defined as the RCPB, includes all pressure-retaining components, such as pressure vessels, piping, pumps, and valves, which are part of the RCS or connected to the RCS. The RCPB includes any components up to and including the following:

- The outermost containment isolation valve in piping that penetrates containment
- The second of the two valves normally closed during normal reactor operation in system piping that does not penetrate containment
- The RCS safety/relief valve (SRV) and safety valve (SV) piping and the depressurization valve (DPV) piping

Section 5.4 of this report addresses various reactor systems. The DPVs are part of the automatic depressurization system (ADS) function of the emergency core cooling system (ECCS) discussed in Section 6.3 of this report. The nuclear boiler system (NBS) pressure relief system protects the RCPB from damage resulting from overpressure. To protect against overpressure, pressure-operated SRVs and SVs discharge steam from the NBS to the suppression pool or to the drywell. The pressure relief system also acts to automatically depressurize the NBS in the event of a loss-of-coolant accident (LOCA) in which the high-pressure makeup of the feedwater (FW), isolation condenser (IC), and control rod drive (CRD) systems fail to maintain the reactor vessel (RV) water level. Depressurization of the NBS by actuation of the SRVs, SVs, and DPVs allows the gravity-driven cooling system (GDCS) to supply cooling water to adequately cool the fuel in the core. Section 5.2.5 of this report specifies the limits on NBS leakage inside the drywell so that operators can take appropriate action to prevent impairment of the integrity of the NBS process barrier.

Section 5.3 of this report describes the RV and appurtenances. The major safety consideration for the RV is its ability to function as a radioactive material barrier. The vessel design considers various combinations of loading. The design process considers the possibility of brittle fracture; addresses suitable design, material selection, and material surveillance activity; and establishes operational limits that avoid conditions in which brittle fracture is possible.

The RCS provides coolant flow through the core by natural circulation within the RV. The core coolant flow rate changes with reactor power output. The control rods are adjusted either manually or automatically with the fine motion CRDs to adjust reactor power. The natural circulation within the RV eliminates the need for a recirculation system. Therefore, there are no large piping connections to the RV below the core, and there are no recirculation pumps.

Venturi-type main steamline (MSL) flow restrictors are part of the main steam nozzle on the reactor pressure vessel (RPV). The restrictors are designed to limit the loss of coolant resulting from an MSL break inside or outside the containment. The restrictors limit the reactor depressurization rate to a value that will ensure that the steam dryer and other reactor internal structures remain in place and limit the radiological release outside of containment before closure of the main steam isolation valves (MSIVs).

Two isolation valves are installed on each MSL. One is located inside the containment and the other is located outside the containment. If an MSL break were to occur inside the containment, closure of the isolation valve outside the containment isolates the containment. The MSIVs automatically isolate the RCPB when a pipe break occurs outside containment. This action limits the loss of coolant and the release of radioactive materials from the NBS.

The CRD system high-pressure makeup provides water by means of the reactor water cleanup/shutdown cooling (RWCU/SDC) piping to the core any time FW flow is unavailable. The high-pressure makeup mode starts automatically upon receipt of a low reactor water level signal; however, the operator can also start it manually. Section 4.6 of this report discusses the CRD system.

The RWCU/SDC system and the isolation condenser system (ICS) can be used to cool the NBS under a variety of situations. During normal shutdown and reactor servicing, the RWCU/SDC system removes residual and decay heat. The RWCU/SDC system, in conjunction with the ICS, allows decay heat to be removed whenever the main heat sink (main condenser) is not available (e.g., hot standby). The ICS provides cooling of the reactor if the RCPB becomes isolated following a scram during power operations.

The ICS automatically removes residual and decay heat to limit reactor pressure when reactor isolation occurs. Over a longer duration, the ICS provides a way to remove excess heat from the reactor with minimal loss of coolant inventory, if the normal heat removal path is unavailable.

The GDCS is an ECCS for use during a postulated LOCA. The GDCS is operational at low RV pressure following pressure reduction by the ADS function of the ECCS. Section 6.3 of this report describes the operation of the GDCS and ADS. The RWCU/SDC system recirculates a portion of reactor coolant through a demineralizer to remove dissolved impurities and their associated corrosion and fission products from the reactor coolant. This system also removes excess coolant from the reactor system under controlled conditions.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with Code and Code Cases

General Design Criterion (GDC) 1, "Quality standards and records," 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," requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement applies to both pressure-retaining and nonpressure-retaining SSCs that are part of the RCPB, as well as to other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability.

5.2.1.1 *Compliance with 10 CFR 50.55a*

5.2.1.1.1 Regulatory Criteria

The staff reviewed Section 5.2.1.1 of the design control document (DCD), Tier 2, in accordance with Section 5.2.1.1, Revision 3, of NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” March 2007 (SRP).

In accordance with 10 CFR 50.55a, components important to safety are subject to the following requirements:

- RCPB components must meet the requirements for Class 1 (Quality Group (QG) A) components, as specified in Division 1, Section III, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2).

- Components classified as QG B and C must meet the requirements for Class 2 and 3 components, respectively, as specified in ASME Code, Section III.

5.2.1.1.2 Summary of Technical Information

DCD Tier 2, Revision 9, Table 3.2-1 classifies the pressure-retaining components of the RCPB as ASME Code, Section III, Class 1 components. These Class 1 components are designated QG A in conformance with Regulatory Guide (RG) 1.26, Revision 3, “Quality Groups Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” issued February 1976. The staff evaluated the QG classifications discussed in Section 3.2.2 of this report and finds that the economic simplified boiling-water reactor (ESBWR) mechanical and pressure-retaining components in the RCPB have been acceptably classified as QG A, in accordance with 10 CFR 50.55a.

In addition to the QG A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B, in accordance with Position C.1 of RG 1.26, and will be constructed as ASME Code, Section III, Class 2 components. Section 3.2.2 of this report discusses the staff’s review of these components and other pressure-retaining components that will be constructed to ASME Code, Section III, Class 2 and 3 specifications.

SRP Section 5.2.1.1 recommends that safety analysis reports for both construction permits and operating licenses contain a table identifying the ASME component code, code edition, and applicable code addenda for all ASME Code, Section III, Class 1 and 2 pressure vessel components, piping, pumps, and valves in the RCPB. DCD Tier 2, Section 5.2.1.1, Revision 9, provides ASME Code edition and applicable addenda for the ESBWR design in compliance with the requirements of 10 CFR 50.55a. DCD Tier 2, Revision 9, Table 1.9-22, identifies the specific ASME Code edition and addenda.

The combined license (COL) applicant must ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda in effect at the time of the COL application, as endorsed in 10 CFR 50.55a. If the ASME Code edition and addenda differ from that specified in the DCD, the COL applicant should identify in its application the portions of the later ASME Code editions and addenda for NRC review and approval.

5.2.1.1.3 Staff Evaluation

ESBWR DCD Tier 2, Revision 5, Section 5.2.1.1, did not address the ASME Code of record edition and addenda used for the design of the ESBWR Class 1, 2, and 3 piping and components. However, DCD Tier 2, Rev. 5, Table 1.9-22, identified the 2001 edition throughout and included the 2003 addenda of the ASME Code as the code of record. The applicant noted in the table that all limitations and modifications specified in 10 CFR 50.55a must be met. However, the staff notes that the 2001 edition throughout and including the 2003 addenda of the ASME Code is excluded from the seismic design for piping by 10 CFR 50.55a(b)(1)(iii). In Request for Additional Information (RAI) 5.2-75, the staff requested that the applicant specify and document an acceptable ASME Code and ASME Code editions and addenda to be used for the design of ESBWR piping and components, in accordance with the requirements of 10 CFR 50.55a. The staff noted that information regarding the ASME Code of record is a Tier 2* information item, requiring NRC approval if the information must be changed in the DCD Tier 2 final safety analysis report (FSAR).

The ASME Code is Tier 1 information; however, the specific edition and addenda are Tier 2* information in part because of the continually evolving design and construction practices (including inspection and examination techniques) of the ASME Code. Fixing a specific edition and addenda during the design certification stage might result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although reference to a specific edition of the ASME Code for the design of ASME Code class components and their supports is suitable for reaching a safety finding during the design certification stage, the construction practices and examination methods of an updated ASME Code that would be effective at the combined license stage must be consistent with the design practices established at the design certification stage.

To avoid this potential inconsistency for the ESBWR pressure-retaining components and their supports, it is appropriate that the ASME Code be specified as Tier 1 information and the specific edition and addenda as Tier 2* information, thereby allowing the COL applicant the option to revise or supplement the referenced ASME Code edition with portions of the later editions and addenda while continuing to ensure consistency between the design and construction practices. This procedure ensures consistency with the latest design, construction, and examination practices.

In response to RAI 5.2-75, the applicant indicated that it would revise Section 5.2.1.1 of the ESBWR DCD to note the use of the ASME Code, Division 1, Section III, 1992 Edition with 1993 Addenda, for seismic design of piping and the use of ASME Code, 1989 Edition with no addenda, for weld-leg dimensions. In addition, the applicant would also revise DCD, Section 5.2.1.1 to reference the ASME Code of record in Table 1.9-22 as it relates to ASME Code, 2001 Edition throughout and including the 2003 Addenda, which is used for the design of components and supports. The applicant noted that DCD Tier 2, Table 1.9-22 and Section 5.2.1.1, would include and designate those ASME Code editions and addenda used for the ASME Code, Section III piping and components in the ESBWR as Tier 2* information requiring NRC approval for changes. The staff finds the ASME Code editions and addenda used for the ESBWR design of ASME Code Class 1, 2, and 3 piping and components, to be in compliance with 10 CFR 50.55a. Therefore, they are acceptable. This was tracked as Confirmatory Item 5.2-75. The staff reviewed Revision 6 of the DCD and finds that the above information has been properly incorporated. Therefore, the confirmatory item is closed.

In RAI 3.12-1, the staff requested that the applicant explain how it will satisfy the requirements of 10 CFR 50.55a(b). Section 3.12.3.1 of this report discusses the resolution of this issue.

By letter dated March 12, 2010, the applicant requested the use of Code Case N-782 for ESBWR design. This Code case is not included in RG 1.84, Revision 34, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III". In accordance with 10 CFR 50.55a(a)(3), the applicant submitted justification requesting NRC approval for the use of this ASME Code case as a proposed alternative to the rules of Section III Subsection NCA-1140 regarding applied code editions and addenda, as is required by 10 CFR 50.55a(c), (d) and (e).

Code Case N-782 provides that the Code edition and addenda endorsed in a design certified or licensed by the regulatory authority may be used for systems and components constructed to ASME Code, Section III requirements. These alternative requirements are in lieu of requirements that base the edition and addenda on the construction permit date. Reference to Code Case N-782 will 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 proposed alternative provides an acceptable level of quality and safety, because the NRC endorses the quality and safety of the ASME Boiler and Pressure Vessel Code editions and addenda at the time of certification of the design to be at an acceptable level. The use of Code Case N-782 facilitates the use of the ASME Code edition and addenda included in the ESBWR design certification. Therefore, Code Case N-782 will provide the same level of quality and safety as was included in the information reviewed for the ESBWR Design Certification. The applicant also indicated that Code Case N-782 is needed so that design specifications and reports using the 2001 Edition through the 2003 Addenda of the Code, approved in the design certification, can be approved for COL applications. Without NRC approval of Code Case N-782, future COL applicants would be required to seek a departure from the certified design. This is a hardship without a compensating increase in the level of quality and safety and could result in a decrease of standardization. The information provided in this letter is generic and applies to all COL applicants referencing the ESBWR design certification.

The staff finds that the applicant has provided adequate justification for use of Code Case N-782 as a proposed alternative to the rules of Section III Subsection NCA-1140 in accordance with requirements of 10 CFR 50.55a(a)(3) and concludes that this Code Case N-782 is therefore, acceptable to be used in ESBWR.

5.2.1.1.4 Conclusions

Based on its review, the staff finds that the ESBWR ASME Code of record, including the editions and addenda, as set forth above in Section 5.2.1.1.3 of this report, complies with 10 CFR 50.55a and is, therefore, acceptable. As a result, the staff finds that the construction of all ASME Code Class 1, 2, and 3 components and their supports will conform to the appropriate ASME Code editions and addenda, as well as the NRC's regulations, and that component quality will be commensurate with the importance of the safety function of all such components and their supports.

5.2.1.2 *Applicable Code Cases*

5.2.1.2.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Revision 9, Section 5.2.1.2, in accordance with SRP Section 5.2.1.2. The staff's acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

- GDC 1, as it relates to the requirement that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed

- 10 CFR 50.55a, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWRs and pressurized-water reactors by requiring conformance with appropriate editions of specified published industry codes and standards

5.2.1.2.2 Summary of Technical Information

DCD Tier 2, Revision 9, Section 5.2.1.2, states that the ESBWR meets the requirements of (1) GDC 1, as it relates to the requirement that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed, and (2) 10 CFR 50.55a, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWRs by requiring conformance with appropriate editions of specified published industry codes and standards. To meet these requirements, the applicant identified in ESBWR DCD Tier 2, Revision 9, Table 5.2-1 various ASME Code cases that are applicable to the component design, construction, and inspection. The staff has either accepted or conditionally accepted all ASME Code cases identified in Table 5.2-1, as discussed in RG 1.84 and RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

5.2.1.2.3 Staff Evaluation

To meet the requirements of GDC 1 and 10 CFR 50.55a, the staff identifies the ASME Code cases in RGs 1.84, 1.147, and 1.192 that may be applied in the construction, inspection, and operation of pressure-retaining ASME Code Class 1, 2, and 3 components. The only ASME Code cases acceptable for use in the design of ASME Code Class 1, 2, and 3, piping systems in the ESBWR are those that are either conditionally or unconditionally approved in RG 1.84 and are in effect at the time of design certification.

DCD Tier 2, Revision 9, Table 5.2-1 identifies specific ASME Code cases that will be applied in the construction of pressure-retaining Class 1, 2, and 3 components covered by ASME Code, Section III. The staff's review of this table is based on the guidelines in RG 1.84, which discusses the applicable ASME Code cases which the NRC has either conditionally or unconditionally endorsed. The staff has endorsed all of the 13 ASME Code cases identified in Table 5.2-1 of the DCD Tier 2, Revision 9, and included in RG 1.84.

In RAI 5.2-32, the staff asked the applicant to provide either annulled ASME Code cases that are not included in DCD Tier 2, Revision 9, Table 5.2-1, or ASME Code cases that are under development, which will potentially be applied in the design and construction of ESBWR pressure-retaining Class 1, 2, and 3 components covered by ASME Code, Section III. In response, the applicant indicated that it is not planning to use any annulled ASME Code cases that are not already included in DCD Tier 2, Table 5.2-1. The applicant is not aware of any ASME Code cases under development that would be needed for the RCPB. In RAI 5.2-33, the staff noted that DCD Tier 2, Rev. 5, Table 5.2-1, lists ASME Code Case N-71-17 for the design and construction of the ESBWR, although the current approved revision is ASME Code Case N-71-18. The staff asked the applicant to justify the differences between the two revisions in the ESBWR design application. The applicant responded that it will correct Table 5.2-1 to indicate the application of ASME Code Case N-71-18. However, the applicant subsequently deleted this ASME Code case from Tier 2 in Revision 3 of the DCD. In response to RAI 5.2-34, the applicant indicated that it will use only those ASME Code cases approved in RG 1.84 for the design of ASME Code Class 1, 2, and 3 components in the RCS. The ASME Code cases that pertain to ASME Code, Section XI, Division 1, as approved in RG 1.147, are used only as they relate to preservice inspection and inservice inspection (PSI/ISI) of ASME Code components. In RAI 5.2-50, the NRC requested that the applicant discuss those ASME Code cases listed in Table 5.2-1 which the NRC has not approved for use (i.e., ASME Code Cases N-634 and N-491-2) and include a basis for their use. RAI 5.2-50 was being tracked as an open item in the SER with open items. The applicant subsequently deleted ASME Code Cases N-634 and N-491-2 in Revision 5 of the ESBWR DCD. The staff finds this acceptable because the applicant no longer lists unapproved ASME Code cases. In response to RAI 5.2-50 S02, the applicant indicated that it will use American Society of Testing and Materials (ASTM) A709 HPS 70W material for containment internal structures. The use of this material falls under ASME Code, Section III, Division 2, and is not applicable to ASME Code cases used for RCPB components. Section 3.8 of this report discusses the ASME Code cases used for ASME Code, Section III, Division 2, applications. RAI 5.2-50 and associated open item are, therefore, resolved.

On the basis of the above evaluation, the staff finds that the applicant will no longer use certain unapproved Code cases as they are deleted in DCD Tier 2, Revision 9, Table 5.2-1. The staff also finds that all of the ASME Code cases listed in DCD Table 5.2-1 meet the guidelines of RG 1.84 in that the staff has reviewed and endorsed these ASME Code cases. The staff finds that the applicant's compliance with the requirements of these ASME Code cases will result in component quality that is commensurate with the importance of the safety functions of the affected components.

5.2.1.2.4 Conclusions

The staff has reviewed the ASME Code cases listed in Table 5.2-1 of ESBWR DCD Tier 2, Revision 9, which meet the guidelines of RGs 1.84, 1.147, and 1.192. The specified ASME and American National Standards Institute (ANSI) Code cases that will apply in the construction of components covered by ASME Code, Section III, Division 1, Class 1, 2, and 3, and Class MC are consistent with the requirements of 10 CFR 50.55a and GDC 1, as well as the guidance provided in RGs 1.84, 1.147, and 1.192. Therefore, the staff considers that the applicant's compliance with the requirements of these ASME Code cases will result in component quality that is commensurate with the importance of the safety functions of the affected components.

5.2.2 Overpressure Protection

5.2.2.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 5.2.2, in accordance with SRP Section 5.2.2, draft Revision 3, "Overpressure Protection."

The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, generic issues (GIs), bulletins (BLs), generic letters (GLs), or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 3 of SRP Section 5.2.2, issued in 1996, is acceptable for this review.

During power operation, SRVs, SVs, and the reactor protection system provide overpressure protection for the RCPB. For the ESBWR, the staff's review covered the SRVs and SVs on the MSLs and piping from these valves to the suppression pool and the drywell.

Acceptance criteria are based on GDC 15, "Reactor coolant system design," and GDC 31, "Fracture prevention of reactor coolant pressure boundary." Specifically, the acceptance criteria are based on GDC 15 as it relates to the design of the RCS and associated auxiliary, control, and protection systems having sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation.

In addition, SRP Section 5.2.2 states that the acceptance criteria are based on GDC 31 as it relates to the fracture behavior of the RCPB. Section 5.2.3 of this report addresses this review area. Overpressure protection during low-temperature operation is not considered for the ESBWR, since there is a very low probability of the ESBWR operating in water-solid conditions. Therefore, this report does not address overpressure protection during low-temperature conditions for the ESBWR.

The ESBWR design must meet the requirements of 10 CFR 50.34(f) which reference Three Mile Island-2 (TMI-2) Action Items II.D.1 "Testing Requirements, II.D.3, "Relief and Safety Valve Position Indication," and II.K.3.16, "Reduction of Challenges and Failures of Relief Valve Feasibility Study and System Modification."

5.2.2.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Sections 5.2.2 and 15.5.1, describe the overpressure protection and the overpressure protection analyses.

The NBS relief system consists of 10 SRVs and 8 SVs located on the MSL between the RPV and the inboard MSIV. The SRVs and the SVs provide the two main protection functions of overpressure protection and automatic depressurization.

The SRVs and SVs function as SVs and open by steam pressure to prevent NBS overpressurization. The safety mode of operation is initiated when direct and increasing static inlet steam pressure overcomes the restraining spring and frictional forces acting against the inlet steam pressure at the valve disc. This moves the disc in the opening direction. The condition at which this actuation is initiated corresponds to the set-pressure value stamped on the nameplate of the valves.

The SRVs and SVs meet the requirements of Section III of the ASME Code. The rated capacity of the SRVs is sufficient to prevent a rise in pressure within the RPV to more than 110 percent of the design pressure during anticipated operational occurrences (AOOs).

Of the 18 total valves, 10 are ADS valves and open automatically during a LOCA to depressurize the RV. The depressurization function is accomplished through the use of SRVs and eight squib-actuated DPVs.

Each SRV has one dedicated, independent pneumatic accumulator, which provides the safety-related nitrogen supply for opening the valve.

The SRVs and SVs are flange mounted onto forged outlet fittings located on the top of the MSL piping in the drywell.

The SRVs and SVs are actuated in groups of valves at staggered times by delay timers as the reactor undergoes a relatively slow depressurization. This minimizes reactor water level swell during the depressurization, thereby enhancing the passive resupply of coolant by the GDCS.

The use of a combination of SRVs, SVs, and DPVs to accomplish the ADS function improves ADS reliability against common-mode failures. Because the SRVs serve two different purposes, overpressure protection and automatic depressurization, the number of required DPVs is minimized. Using DPVs for the additional depressurization capability needed beyond what the SRVs can provide minimizes the total number of SRVs, SRV discharge lines, and quenchers in the suppression pool. This arrangement also minimizes the need for SRV maintenance and periodic calibration and testing, as well as the potential for simmering.

The ADS automatically actuates on a low RPV water-level signal that persists for a preset time. Two-out-of-four logic is used to activate the SRVs and DPVs. The persistence requirement for the low RPV water-level signal ensures that momentary system perturbations do not actuate the ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation, while also ensuring that a single failure cannot prevent initiation. The ADS may also be manually initiated from the main control room (MCR).

5.2.2.3 *Staff Evaluation*

The staff assessed the design and function of the ESBWR overpressure protection system.

The pressure relief system for the RCPB does the following:

- Prevents the pressure in the RCPB from rising beyond 110 percent of the design value

- Provides automatic depressurization for breaks in the NBS so that the GDCS can operate to protect the fuel barrier

An earlier version of the DCD Tier 2, Section 5.2.2.1, stated, “the rated capacity of the pressure relieving devices shall be sufficient so that the rise in pressure within the protected vessel does not exceed 120 percent of the design pressure for pressurization events described in Chapter 15.” Based on a review of Chapter 15, the staff’s position was that the pressure limit for pressurization events is 110 percent of the design pressure. Therefore, the staff required the applicant to revise the DCD to use 110 percent as a pressure limit instead of 120 percent. The staff tracked RAIs 15.0-17 and 15.0-18 as open items in the SER with open items. The applicant revised the DCD Section 5.2.2 to state that the RPV will not exceed 110 percent of the design pressure and the staff verified that this change was incorporated into the DCD accordingly. Therefore, RAIs 15.0-17 and 15.0-18 are resolved

For the pressure relief system to be acceptable, it must be possible to verify its operability and its ability to withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions. Section 3.6 of this report evaluates protection against the dynamic effects associated with the postulated rupture of piping.

For overpressure protection, the ICs have sufficient capacity to preclude actuation of the SRVs during normal operational transients. The SRVs limit the pressure to less than the RCPB design pressure during more severe transients.

Ten SRVs and 8 SVs provide overpressure protection in the ESBWR. The nominal pressure setpoint of the 10 SRVs is 8.618 megapascals (MPa) (1,250 pounds per square inch gage [psig]), and the nominal set pressure for the 8 SVs is 8.756 MPa (1,270 psig). The SRVs and SVs are mounted on the four MSLs between the RV and the first isolation valve inside the drywell. Ten ADS SRVs discharge through piping to the suppression pool, and eight non-ADS SVs discharge into the drywell. Short discharge pipes with end-mounted rupture disks limit SV discharge from entering the drywell atmosphere during normal operation. Two vacuum relief valves on each SRV discharge line minimize the initial rise of water in the discharge piping. ADS SRVs are provided with nitrogen accumulators and check valves. These accumulators ensure that the valves can be opened following the failure of the normal gas supply. The accumulator capacity is sufficient for one actuation at drywell design pressure.

The SRVs and SVs are classified as QG A and seismic Category I, as shown in DCD Tier 2, Revision 9, Table 3.2-1. The design of the SRVs and SVs is consistent with the guidance in RGs 1.26 and 1.29, Revision 3, “Seismic Design Classification,” issued September 1978.

The ADS SRVs can also be operated in the relief mode by remote-manual controls from the MCR.

GDC 15 defines the basis for overpressurization protection in a nuclear reactor. It requires that the RCPB design conditions not be exceeded during any condition of normal operation, including AOOs. To satisfy this criterion, the overpressurization protection system for the ESBWR is designed to comply with ASME Code, Section III, which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For the ESBWR, that pressure limit is 9.48 MPa (1,375 psig). The applicant used the computer simulation model TRACG to analyze a series of transients that would be expected to require SRV actuation to prevent overpressurization. The GE-Hitachi (GEH) Topical Report, “TRACG Model Description,” NEDE-32176P, Revision 4, issued January 2008, describes the TRACG model. Section 21.6 of this report provides the staff’s evaluation of the TRACG model.

The staff reviewed the overpressure analyses presented in Section 15.5.1 of the DCD and found that the applicant’s assumptions are consistent with the assumptions given in SRP Section 5.2.2. For the most severe transient (i.e., closure of all MSIVs with a high neutron flux scram), the maximum vessel bottom pressure is calculated to be less than the acceptance limit of 9.48 MPa gage (1,375 psig). The analysis assumed that the plant was operating at a rated steam flow of 2,433 kilograms per second (kg/s) (19.3180 million pounds per hour [Mlbm/hr]) and a vessel dome pressure of 7.17 MPa (1,040 psig). The analysis credits the spring action safety mode of only one valve. The ESBWR RPV is larger than that in the currently operating BWRs, and therefore, the reactor pressurization is slower. In general, RPV pressure ceases to increase once a single relief valve opens because of the higher steam volume-to-power ratio of the ESBWR, which causes the pressure increase rate before a scram to be much lower than currently operating BWRs. After a scram, the pressure increase rates resulting from stored energy release are correspondingly lower.

The applicant based the sizing of the SRVs on the initiation of a reactor scram by the high neutron flux scram, which is the second safety-grade scram signal from the reactor protection system following MSIV closure. The staff notes that the spring action mode of only one valve is required for reactor overpressure protection, however, all of the 18 valves are required for an anticipated transient without scram (ATWS), as described in Section 15.5.4 of this report. The staff believes that the qualification and redundancy of reactor protection system equipment, coupled with the limitation of the RPV to less than 110 percent of design pressure, provide adequate assurance that the RV integrity will be maintained for the limiting transient event.

As required by 10 CFR 50.34(f)(1)(vi), which references TMI-2 Action Item II.K.3.16, a study must be performed to identify practical system modifications that would reduce challenges and failures of relief valves in BWRs, without compromising the performance of the valves or other systems. The SRVs are expected to open in the event of an ATWS or the occurrence of beyond design-basis events. However, one of the key design criteria of the ESBWR is that SRVs shall not need to open during most transients to protect against overpressure. Rather, overpressure protection is achieved through the use of the ICS. GEH and the Boiling-Water Reactor Owners Group (BWROG) responded to this requirement for their earlier boiling-water reactor (BWR) models. Based on a review of the existing operating information on the challenge rate of relief valves, BWROG concluded that the BWR/6 product line had already achieved a level of reduction in the SRV challenge rate. The principal reason for this reduction is that the BWR/6 uses direct-acting SRVs, rather than the pilot-operated design used in some earlier BWRs. The ESBWR uses a modern and improved SRV and SV design; therefore, earlier problems are not expected to occur. The staff finds that the ESBWR design complies with 10 CFR 50.34(f)(1)(vi) and TMI-2 Action Item II.K.3.16.

In Revision 2 of the DCD, the applicant deleted the following statement found in DCD, Revision 1, Section 5.2.6, without including a reason for the deletion:

> The COL applicant is required to submit an overpressure protection analysis for core loadings different than the reference ESBWR core loading.

Rather than deleting this sentence, the staff believed that the applicant should have revised it to state the following:

> The COL applicant is required to submit an overpressure protection analysis for the actual core for the initial startup.

The staff tracked RAI 5.2-61 as an open item in the SER with open items. In response to RAI 5.2-61, the applicant stated that the overpressure protection analysis for the initial core is included in Topical Report NEDO-33337, “ESBWR Initial Core Transient Analyses,” issued October 2007. The analysis with only one safety relief valve showed that there is sufficient margin for the reactor overpressure protection. If credit is given to all the ten safety relief valves in the analysis, it is expected that there will be significant margin for overpressure protection. Moreover, the pressurization transients in the ESBWR are expected to be less severe than in current operating BWRs, therefore the staff decided that the applicant need not submit cycle-specific overpressure protection analyses for staff review. Therefore, RAI 5.2-61 and associated open item are resolved.

As required by 10 CFR 50.34(f)(2)(x), which references TMI-2 Action Item II.D.1, licensees must provide a test program with associated model development and conduct tests to qualify RCS relief and SVs for all fluid conditions expected under operating conditions, transients, and accidents. The test program must consider ATWS conditions. For currently operating plants, a generic test program for current valve designs and plant-specific responses for individual plant piping configurations and system responses resolved this issue. The applicant must either confirm that the generic test program for currently operating plants is applicable to ESBWR transients and accidents or commit to perform the required testing and provide necessary plant-specific testing. In ESBWR DCD Tier 2, Revision 9, Table 1A-1, the applicant stated that the SRVs will be tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability before installation. The tests will include hydrostatic, steam leakage, full-flow pressure and blowdown, and response time testing. The valves will be installed as received from the factory. The valve manufacturer will certify that design and performance requirements, including capacity and blowdown, have been met. The vendor will adjust, verify, and indicate the setpoints on the valves. Specified manual and automatic initiation signals for power actuation of each ADS SRV will be verified during the preoperational test program described in Chapter 14 of the DCD. The applicant also stated that the inspection and test program for the SRVs will follow a quality assurance program that complies with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. SRV setpoints will not be tested in place, but the SRVs will be removed for maintenance or bench testing and reinstalled during normal plant shutdowns. The valves will be tested to check set pressure in accordance with the requirements of the plant's technical specifications (TS). Further, as discussed in Section 3.9.3 of this report, the external and flange seating surfaces of the SRVs will be visually inspected when the valve is removed for maintenance or bench testing. The staff finds these actions to be consistent with the applicable TMI Action Item II.D.1 provision and to address lessons learned from SRV performance at operating nuclear power plants. Therefore, as discussed in Section 3.9.3 of this report, the staff finds the applicant's response to be acceptable and TMI Action Item II.D.1 is resolved.

In accordance with the requirements of 10 CFR 50.34(f)(2)(xi), which references TMI-2 Action Item II.D.3, the control room includes SRV and SV position indications.

SRV setpoint drift and seat leakage are generic problems. In the response to RAI 5.2-20, the applicant addressed (1) specific design features of the ESBWR SRVs, (2) a comparison of the relative performance of ESBWR SRVs and SRVs currently installed in operating reactors, and (3) a detailed description of any improvements between the ESBWR SRV design and the design of SRVs presently installed in operating reactors in terms of seat leakage, setpoint drift, and actuator reliability.

In the response to RAI 5.2-21, the applicant addressed (1) improvements in the air actuator, especially materials used for components such as diaphragms and seals, (2) safety margins associated with the air accumulator design, (3) pressure indications in the accumulator and how this information is relayed to the operator, and (4) provisions employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (i.e., radiation, temperature, humidity, and vibration).

In the response to RAI 5.2-20, the applicant stated that it had not finalized the detailed design and selection of the ESBWR SRVs. In the response to RAI 5.2-22, the applicant stated that it will prepare a purchase specification for the SRVs, which uses the applicant's environmental qualification experience base. The SRVs will be subject to the environmental and dynamic qualification program. In the response to RAI 5.2-7, the applicant stated that, consistent with past practice, it will prepare a purchase specification for the SRVs, which addresses the inspection and test requirements of the program. In regard to RAIs 5.2-7, 5.2-20, and 5.2-22, the staff requested that the applicant specify its acceptance criteria for the design and qualification of the SRVs to be used in the ESBWR, including appropriate inspection, test, analysis, and acceptance criteria (ITAAC). The applicant responded that Item 1 in DCD Tier 1, Revision 3, Table 2.1.2-2, contains an ITAAC to confirm the basic configuration for the NBS and states that those inspections must be conducted using the acceptance criteria that the as-built NBS conforms to the basic configuration, as defined in DCD Tier 1, Section 2.1.2. The applicant believes that this ITAAC includes programmatic reviews of SRV design and environmental qualifications which meet the intent of the supplemental RAI in which the staff requested that the applicant specify its acceptance criteria for the design and qualification of the ESBWR SRVs, including appropriate ITAAC.

DCD Tier 1, Revision 3, Section 1.1.12.2.1(4), states that the basic configuration ITAAC includes the following:

> Type tests or type tests and/or analyses, of the safety related mechanical equipment demonstrate qualification to applicable normal, abnormal and design basis accident conditions with out loss of the safety-related function for the time needed during and following the conditions to perform the safety related function considering the applicable harsh environmental conditions.

The staff responded with the following supplemental request:

A. The referenced ITAAC is not sufficient. Revise the ITAAC table to include verification for the SRV discharge capacity and set points to demonstrate that the as-built is consistent with the assumptions of the safety analyses.

B. Include a COL Applicant or COL Holder Item to the DCD to ensure that operating experience, for example, issues identified in Regulatory Issue Summary 00-012, “Resolution of Generic Safety Issue B-55, ’Improved Reliability of Target Rock Safety Relief Valves,’” Inspection and Enforcement Office (IE) Circular 79-18, “Proper Installation of Target Rock Safety Relief Valves,” BL 74-04, “Malfunction of Target Rock Safety Relief Valves,” and NUREG-0763, “Guidelines for Confirmatory In-plant Tests of Safety Relieve Valve Discharges for BWR Plants” are addressed when the SRVs are procured.

C. Revise the DCD Tier 1, Section 1.2.2.1 to expand the environmental qualification verifications to include mechanical equipment such as seals and gaskets.

The staff tracked RAIs 5.2-20 and 5.2-22 as open item in the SER with open items.

In response to the staff’s concern regarding generic problems with pilot-operated SRVs (e.g., setpoint drift, seat leakage), the applicant stated in response to RAI 5.2-20 S03, that it will consider operating experience when selecting the design of the SRVs and SVs. The following generic communications will be factored into the selection of SRVs and SVs:

- Generic Safety Issue B-55, “Improved Reliability of Target Rock Safety Relief Valves.”

- NUREG–0763, “Guidelines for Confirmatory In-Plant Tests of Safety Relief Valve Discharges for BWR Plants.”

- IE Circular 79-18, “Proper Installation of Target Safety Relief Valves.”

- BL 74-04, “Malfunction of Target Rock Safety Relief Valves.”

The applicant indicated that the ESBWR will use either direct-acting SRVs or a modern pilot-operated SRV design that has been proven not to experience the performance problems observed in earlier BWRs. As discussed in Section 3.9.6 of this report, since the applicant agreed to consider the operating experience when selecting the design of the SRVs, the applicant’s response is acceptable.

In response to RAI 5.2-22, the applicant included SRV and SV discharge capacity and setpoints verification in the ITAAC 2.1.2, “Nuclear Boiler System”. The staff will verify this during the ITAAC closure phase.

The applicant provided necessary information requested by the staff; therefore, RAIs 5.2-20 and 5.2-22 and associated open items are resolved.

Operating experience has shown that SRV failure may be caused by exceeding the manufacturer’s recommended service life for the internals of the SRV or air actuator. In addition to periodic testing, the licensee shall perform valve inspection and overhaul in accordance with the manufacturer’s recommendations. In response to RAI 5.2-25, the applicant stated, “Every 5 years during reactor plant shutdown, the valves are subjected to a complete visual examination, set pressure testing and seat tightness testing.” The licensee will test SRVs in accordance with the inservice testing (IST) program as discussed in Section 3.9.6 of this report.

The effects of flow-induced SRV discharge line back pressure on the performance of the SRV are addressed by sizing the line to ensure that the steady-state back pressure does not exceed 40 percent of the SRV inlet pressure. This sizing criterion controls the effective back-pressure buildup and maintains the required force balance needed to keep the SRV open and permit proper blow down. The non-ADS SRVs discharge through the rupture discs to the drywell. In response to RAI 5.2-10, the applicant stated that the design of the rupture disc will comply with ASME Code, Subsection NB-7623.

Before the valves are installed, the SRV manufacturer will test the valves hydrostatically according to the requirements of ASME Code, Section III. During startup testing, opening response time and set-pressure tests will be conducted to verify that design and performance requirements have been met.

5.2.2.4 *Conclusions*

For the reasons set forth above, staff finds that the pressure relief system, in conjunction with the ICS and the reactor protection system will provide adequate protection against overpressurization of the RCPB. The staff further finds that the overpressurization system is acceptable and meets the relevant requirements of GDC 15.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 5.2.3, in accordance with SRP Section 5.2.3, Revision 3. The materials specifications, compatibility of materials with the reactor coolant, fabrication and processing of ferritic materials, and fabrication and processing of austenitic stainless steel within the RCPB are acceptable if they meet the relevant requirements set forth in 10 CFR 50.55a; GDC 1, 4, “Environmental and dynamic effects design bases,” 14, “Reactor coolant pressure boundary,” 30, “Quality of reactor coolant pressure boundary,” and 31; Appendix B to 10 CFR Part 50; and Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50. These requirements are discussed below:

- Compliance with GDC 1 and 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- Compliance with GDC 4 requires that 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 LOCAs.

- Compliance with GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

- Compliance with GDC 30 requires that components of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

- Compliance with GDC 31 requires that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

- Compliance with Appendix B to 10 CFR Part 50 requires, in Criterion XIII, that measures be established to control the cleaning of material and equipment to prevent damage or deterioration.

- Compliance with Appendix G to 10 CFR Part 50 requires that the fracture toughness of RCPB ferritic materials be tested in accordance with the requirements of the ASME Code and that the pressure-retaining components of the RCPB that are made of ferritic materials meet requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs.

5.2.3.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Table 5.2-4, lists the principal pressure-retaining materials and material specifications for the RCPB components. This list includes the MSIVs, SRVs and DPVs, main steam piping, CRD components, RPV, IC piping, and FW piping.

The materials used in the RCPB, including materials that do not act as a pressure boundary, consist of austenitic wrought and cast stainless steel, nickel-based alloys, carbon and low-alloy steels, 400 series martensitic stainless steel, Colmonoy and Stellite hard-facing alloys, and precipitation-hardened stainless steels. The applicant indicated that it considered the compatibility of the materials of construction used in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed in the ESBWR design. All of the construction materials are resistant to stress-corrosion cracking (SCC) in the BWR environment. General corrosion of all materials, with the exception of carbon and low-alloy steel, is negligible. The applicant considered the extent of the corrosion of ferritic low-alloy steels and carbon steels in contact with the reactor coolant in the design by providing corrosion allowance for all exposed carbon steel and low alloy steel surfaces.

The ESBWR design complies with RG 1.44, “Control of the Use of Sensitized Stainless Steel,” issued May 1973; RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” issued February 1973; GL 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,” dated January 25, 1988; and NUREG–0313, Revision 2, “Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,” issued January 1988.

Fracture toughness for ASME Code Class 1 ferritic materials used for the reactor piping, pumps, and valves meets the impact testing requirements of ASME Code, Subsections NB-2331 and NB-2332. Materials for bolting meet the requirements specified in Subsection NB-2333.

The ESBWR design provides alternatives to the guidelines in RG 1.50, “Control of Preheat Temperature Employed for Welding of Low-Alloy Steel,” issued May 1973, and RG 1.71, Revision 1, “Welder Qualification for Areas of Limited Accessibility,” issued March 2007.

Wrought tubular products that are used for pressure-retaining components of the RCPB are subject to the examination requirements of ASME Code, Section III, Subsection NB.

These RCPB components meet the requirements of Appendix B to 10 CFR Part 50 and the ASME Code, thus ensuring adequate control of product quality.

5.2.3.3 *Staff Evaluation*

As discussed below, the staff evaluated material specifications, compatibility of the materials with the reactor coolant, fabrication and processing of ferritic materials and fabrication, and processing of austenitic stainless steel.

5.2.3.3.1 Material Specifications

The specifications for pressure-retaining ferritic materials, nonferrous metals, and austenitic stainless steels, including weld materials that are used for each component in the RCPB, must meet the requirements of GDC 1 and 30 and 10 CFR 50.55a, as they relate to quality standards for design, fabrication, erection, and testing. These requirements are met for material specifications by complying with the appropriate provisions of the ASME Code, by applying the ASME Code cases identified in RG 1.84, and by complying with the guidelines of NUREG–0313, Revision 2.

The staff reviewed DCD Tier 2, Section 5.2.3.1, to determine the suitability of the RCPB materials for this application. The staff determined that the applicant’s material specifications listed in DCD Tier 2, Section 5.2.3 and Table 5.2-4, for the ESBWR design conform with the guidance in RG 1.84 and NUREG–0313, Revision 2, as well as the appropriate provisions of the ASME Code and other staff guidance except as noted below.

Adhering to the guidance provided in NUREG–0313, Revision 2, appropriately addresses GL 81-03, “Implementation of NUREG–0313, Technical Report on Material Selection & Processing GL for BWR Coolant Press Boundary Piping,” dated February 26, 1981, and GL 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping.” In addition, NUREG–0933, “A Prioritization of Generic Safety Issues,” Task Action Plan Issue A-42 related to pipe cracks in BWRs, New GI 119.4 related to the use of materials resistant to intergranular stress-corrosion cracking (IGSCC), and New GI 86 related to the long-range plan for dealing with SCC in BWR piping are resolved for the ESBWR design based on the applicant's adherence to the guidance provided in NUREG–0313, Revision 2.

The staff noted that DCD Tier 2, Table 5.2-4 did not include the material specifications and grades of some major components. The table did not include isolation valves and check valves in the ASME Code Class 1 portion of the FW piping. Table 5.2-4 must include the material specifications for these components. The staff tracked this issue as open item (RAI 5.2-36) in the SER with open items. The staff subsequently reviewed the applicant's modifications to Table 5.2-4 in Revision 5 of the DCD. The applicant modified Table 5.2-4 to include material specifications for FW valves in the RCPB. During its review of modifications to Table 5.2-4, the staff identified portions of the table that required clarification and issued RAI 5.2-36 S02, to resolve the issues listed below.

For DPV bodies, FW valves, and FW disc, the applicant listed SA-426, "Specification for Centrifugally Cast Ferritic Alloy Steel Pipe for High Temperature Service," Grade CP22. The staff requested that the applicant explain why it selected a cast pipe specification for valve bodies and valve disc. The staff also noted that the Table 5.2-4 references to SA-376 TP 304L and 316L must be listed as TP 304LN and TP 316LN to be consistent with SA-376 and ASME Code, Section III, Part D.

The applicant listed SFA-5.23 FS8PO-ECF2-F2H2 under welding filler metals for P3, Group 3 base materials. The staff noted that this classification specifies that the welding flux is made solely from crushed slag or is a blend of crushed slag with unused flux. The staff requested that the applicant explain how a consistent weld metal chemistry will be maintained using crushed slag or a combination of crushed slag and unused flux rather than unused flux alone. In addition, neither of the submerged arc welding specifications listed in Table 5.2-4 contains an "N" designator for special requirements related weld filler metal used in the core beltline. In RAI 5.2-36 S02, the staff requested the applicant to delete "Note 4" of Table 5.2-4, which indicates that filler materials listed in Table 5.2-4 are representative and may be changed.

The applicant responded and provided a proposed revision to Table 5.2-4 to address the staff's concerns identified in RAI 5.2-36 S02. The applicant's proposed revision to Table 5.2-4 deleted specification SA-426 Grade CP22 and replaced it with specification SA-217 Grade WC9. SA-217 is an appropriate specification for the fabrication of valves and Grade WC9 is a low-alloy steel which matches the flow-accelerated corrosion resistance of the FW piping material specified by the applicant. This material is also listed in ASME Code, Section II, Part D, as an acceptable material for use in Class 1 systems. Therefore, the staff finds this material acceptable.

The applicant's revision to Table 5.2-4 also changed SA-376 TP 304L and 316L to TP 304LN and TP 316LN which the staff finds acceptable because the applicant's reference to the above material grades is now consistent with SA-376 and ASME Code, Section III, Part D. In addition, the applicant's revised Table 5.2-4 deletes the use of weld filler metal that includes recycled flux. The staff finds this acceptable because the use of recycled flux could adversely affect final weld metal chemistry. The applicant also modified Table 5.2-4 to add Notes 5 and 6 to address special weld filler metal requirements for the core beltline. Note 5 provides a reference to DCD Table 5.3-1 and Section 5.3.1.5. DCD Table 5.3-1 and Section 5.3.1.5 specify core beltline composition limits that meet or exceed the requirements of SFA-5.23 for weld material classifications that use the "N" designator. DCD Table 5.2-4, Note 6 requires additional impact testing for core belt line materials per SFA-5.23 “N” designation. The staff finds this acceptable because the core beltline weld materials will meet all applicable requirements of SFA 5.23. The staff subsequently reviewed ESBWR DCD, Revision 6, and verified that the applicant made the modifications to the DCD discussed above. RAI 5.2-36 S02 and its associated open item are therefore resolved.

DCD, Section 3E.2.2 listed SA-672 Grade C70 as a material used in the RCPB. However, this material was not listed in Table 5.2-4. The staff requested, in RAI 5.2-37, that the applicant correct this inconsistency. In the applicant’s response to RAI 5.2-37, it indicated that SA-672 Grade C70 was listed in error. The applicant provided a proposed revision to Section 3E.2.2 that listed SA-106 Grade B and SA-333 Grade 6. The staff noted that SA-106 Grade B was not listed in Table 5.2-4. In RAI 5.2-37 S01, the staff requested that the applicant correct this inconsistency. The staff tracked RAI 5.2-37 as an open item in the SER with open items. In the applicant’s response to RAI 5.2-37 S01, it indicated that it did not intend to use material specification SA-106 and that this material would be deleted form the DCD for use in the RCPB. The staff reviewed DCD, Revision 5 and verified that the applicant removed references to SA-106 for use in the RCPB. RAI 5.2-37 and the associated open item are resolved.

DCD Tier 2, Table 5.2-4, indicated that the RCPB includes cast austenitic stainless steel (CASS) components. CASS components used in light-water reactors (LWRs) can be susceptible to thermal aging embrittlement. In RAI 5.2-38, the staff asked the applicant to provide the following information for any CASS component that is part of the RCPB: (1) the impact of this aging effect on the integrity of the components, (2) the consideration of the thermal embrittlement mechanism in the design and material selection for RCPB components, (3) the need for inspections to detect this aging effect, and (4) verification that the δ-ferrite content is calculated using Hull’s equivalent factors or a method producing an equivalent level of accuracy. In response, the applicant referenced its response to RAI 4.5-3, for Items 1, 2, and 3 above. The applicant stated that, at the normal operating temperature for all BWRs of 550 degrees Fahrenheit (F), thermal aging of low carbon stainless steel castings with less than 20-percent ferrite is barely measurable. The applicant also stated that these materials have more than 35 years of operating experience with no problems or failures.

The applicant’s responses to Items 1, 2, and 3 above meet the staff’s expectation that the applicant screening process for the determination of CASS susceptibility to thermal aging embrittlement, for materials listed in Table 5.2-4, is consistent with the staff position documented in a letter from Christopher I. Grimes of the NRC to Douglas J. Walters of the Nuclear Energy Institute, dated May 19, 2000 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML003717179). For Item 4, the applicant stated that it intends to use ASTM A800 to determine δ-ferrite content in lieu of Hull’s equivalent factors. This is inconsistent with the staff’s position that ferrite content be calculated using Hull’s equivalent factors as indicated in NUREG/CR–4513, Revision 1, “Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems,” issued May 1994. For ferrite content above 12 percent, ASTM A800 may produce nonconservative ferrite levels lower than those calculated using Hull’s equivalent factors. In response to RAI 6.1-15, the applicant stated that use of a rigorous statistical analysis can demonstrate that the two methods are equally accurate. In RAI 5.2-38 S01, the staff requested the applicant to provide a rigorous statistical analysis showing that the method to calculate ferrite using ASTM A800 and the method using Hull’s equivalent factors are equally accurate. The applicant provided its statistical analysis in response to RAI 5.2-38 S01. The staff reviewed it and found it to be unacceptable because it did not show that ASTM A800 and Hull’s equivalent factors are equally accurate. The staff tracked RAI 5.2-38 as an open item in the SER with open items. In response to supplemental RAIs associated with this open item, the applicant informed the staff that it would modify DCD Tier 2, Section 5.2.3.4 to state that, for CASS material used as part of the RCPB or RV internals, the percent ferrite is to be calculated using Hull's equivalent factors, as indicated in NUREG/CR–4513, Revision 1. The applicant also stated that it would modify DCD Tier 2, Section 5.2.3.4 to limit the percent ferrite in CASS material to a maximum value of 20 percent, which is consistent with the staff's position regarding the control of thermal embrittlement in CASS materials. The staff reviewed DCD, Revision 5, and verified that the applicant had completed the aforementioned DCD modifications. RAI 5.2-38 and the associated open item are resolved.

Several operating experience issues have arisen related to the fabrication quality and inservice performance of dissimilar metal welds (DMWs) in LWRs. In RAI 5.2-40, the staff asked the applicant to describe DMWs in the RCPB and discuss the selection of filler metals, welding processes, and process controls for DMWs in the ESBWR design.

In response, the applicant indicated that DMWs are primarily used in the RCPB to join carbon steel to stainless steel piping components. These joints are generally made by applying a buttering layer or layers of 309L or 309MoL followed by completion of the groove weld using 308L, 316L, 309L, or 309MoL. Ferrite content in welds is controlled to between 8FN and 20FN. Postweld heat treatment of the carbon steel after buttering is performed if dictated by ASME Code, Subsection NB-4600. DMWs may also be made using Alloy 82 when welding nickel alloys to carbon steel, low-alloy steel, and stainless steel. Postweld heat treatment of stainless steel components will not be allowed. The staff notes that the aforementioned weld filler materials are considered Category A materials, in accordance with NUREG–0313, and provide an increased level of resistance to IGSCC when compared to non-low-carbon stainless steel welding filler materials. The applicant stated that all of the aforementioned alloys are currently in BWR service with no observed incidences of SCC or other problems, which provides additional assurance that the welds will maintain structural integrity throughout the design life of the plant. Based on the above, the staff finds that the applicant's proposed welding methods and selection of weld filler materials for DMWs are acceptable, because the applicant will follow current industry practice; the weld filler materials are consistent with staff guidance, and these materials have had favorable operating experience.

ASME Code, Section III, Subsection NB-3121, requires that material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects must provide for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas. In DCD Tier 2, Section 5.2.3.2, the applicant indicated that it considered the extent of the corrosion of ferritic low-alloy steels and carbon steels in contact with the reactor coolant in the design. In RAI 5.2 43, the staff asked the applicant to describe corrosion allowances for all unclad low-alloy and carbon steel surfaces in the RCPB. In response, the applicant stated that corrosion allowances for unclad carbon and low-alloy steels are defined for both external (air exposure) and internal (wetted) surfaces. The 60-year allowance for external surfaces is 0.8 millimeters (mm) (0.03 in.) and the allowance for internal surfaces is 1.6 mm (0.06 in.). In response to RAI 6.1-7, the applicant provided additional information regarding its process for determining the corrosion allowance for RCPB ferritic materials. The corrosion allowance is primarily based on the applicant's internal testing. The allowances consider fluid velocity, oxygen content, and temperature and include a safety margin over the actual measured corrosion rates of approximately a factor of 2. The same method, with corresponding allowances, has been applied to most operating BWRs of GEH design, including the advanced boiling-water reactor (ABWR) design. The staff finds this acceptable, given that the applicant has considered the effect of corrosion, based on laboratory testing and operational experience, over the design life of the plant as required by ASME Code, Section III.

DCD Tier 2, Table 5.2-4, indicates that E9018-B3L and ER90S-B3L will be used to weld components in the RCPB. The staff notes that ASME discontinued the aforementioned weld filler metal classifications several years ago and replaced them with classifications E8018-B3L and ER80S-B3L. The same issue exists in the applicant's proposed revision of Table 6.1-1. In RAI 6.1-2 S02, the staff requested the applicant to modify Tables 5.2-4 and 6.1-1 to include the correct weld filler material classifications. Table 5.2-4 and the applicant's proposed version of Table 6.1-1 list the weld filler material that will be used to weld P5C, G1 materials. After reviewing the RCPB and engineered safety feature (ESF) material specifications provided in the DCD and the applicant's response, the staff is unable to identify any materials that fall into the P5C, G1 category, in accordance with ASME Code, Section IX, Table QW/QB-422. In RAI 6.1-2 S02, the staff requested the applicant to identify the P5C, G1 materials used in the ESBWR design for RCPB and ESF components or delete this information from the DCD if it does not apply.

Table 5.2-4 and the applicant's proposed revision to Table 6.1-1 identify shielded manual arc welding filler material E8018-G for use in welding low-alloy steel in the ESBWR design. To complete its review and evaluate the applicant's compliance with 10 CFR 50.55a, the staff requested, in RAI 6.1-2 S02, that the applicant provide the complete GEH specification that will be used to purchase E8018-G for the fabrication of ASME Code Class 1, 2, and 3 components. In addition, the staff asked that the applicant provide a technical justification for using the GEH specification in lieu of commercially available welding electrodes. The staff tracked RAI 6.1-2 as an open item in the SER with open items.

The applicant responded and indicated that it would modify Tables 6.1-1 and 5.2-4 to delete obsolete filler material classifications, delete references to P5C, G1 materials, and delete E8018-G filler material classifications. The staff reviewed Revision 5 to the ESBWR DCD and verified that the applicant had made the appropriate modifications. RAI 6.1-2 and the associated open item are resolved.

The staff finds that the applicant's selection of materials for use in the RCPB meets the requirements of the ASME Code or the guidance of RG 1.84 and complies with the guidelines of NUREG–0313, Revision 2, and is therefore acceptable.

5.2.3.3.2 Compatibility of Materials with the Reactor Coolant

The RCPB materials of construction that are in contact with the reactor coolant, contaminants, or radiolytic products must be compatible and must meet the requirements of GDC 4, as they relate to the compatibility of components with environmental conditions. The applicant stated that it considered the compatibility of the materials of construction used in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the RCPB is exposed.

The applicant stated that the materials of construction are compatible with primary coolant water, which is chemically controlled in accordance with the appropriate TS, as discussed in Section 5.4.8 of this report. The applicant's selection of materials and control of water chemistry will ensure compatibility. Additionally, extensive testing and satisfactory performance of these materials in operating plants for several years have proven this compatibility. The materials meet the requirements of GDC 4 because the ESBWR design complies with the applicable provisions of the ASME Code, adheres to the guidance provided in RG 1.44, and conforms to the staff positions of GL 88-01, which are based on the technical information and recommendations provided in NUREG–0313. Therefore, material compatibility with primary water coolant will be assured.

5.2.3.3.3 Fabrication and Processing of Ferritic Materials

The fracture toughness of ferritic materials in the RCPB must meet the requirements of Appendix G to 10 CFR Part 50. These criteria satisfy the requirements of GDC 14 and 31 regarding prevention of fracture of the RCPB.

Appendix G to 10 CFR Part 50 requires the pressure-retaining components of the RCPB to be made of ferritic materials to meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs. For piping, pumps, and valves, this requirement is met through compliance with the requirements of ASME Code, Section III, Subsection NB-2331 or Subsection NB-2332, and the C_v values specified in Table NB-2332(a)-1. Materials for bolting must meet the impact test requirements of ASME Code, Section III, Subsection NB-2333. Calibration of temperature instruments and C_v impact test machines must meet the requirements of ASME Code, Section III, Subsection NB-2360. The staff reviewed DCD Tier 2, Section 5.2.3.3.1, and verified that the ESBWR design meets the aforementioned requirements regarding fracture toughness of RCPB piping, components, and bolting and equipment calibration. Section 5.3 of this report presents the staff's evaluation of the fracture toughness requirements of the RPV.

Control of ferritic steel welding by following NRC RGs and adhering to the ASME Code satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a. Adherence to the guidance provided in RG 1.50; RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," issued May 1973; RG 1.34, "Control of Electroslag Weld Properties," issued December 1972; RG 1.71; and ASME Code nonmandatory Appendix D, "Non-mandatory Preheat Procedures," Subsection D-1210, satisfies the aforementioned quality standard requirements.

DCD Tier 2, Section 5.2.3.3.2, Revision 9 discusses the use of RG 1.50 and preheat requirements when welding low-alloy steel in the ESBWR design. Low-alloy steel is used only in the RPV and FW piping. The applicant indicated that an alternative to RG 1.50 may be applied to the RCPB components. RG 1.50 provides guidance that all low-alloy steel welds be maintained at the minimum preheat temperature until postweld heat treatment is performed. In RAI 5.2-44, the staff asked the applicant to describe the portions of RG 1.50 that will not be followed and the steps that it will take to ensure that delayed cracking of the weld metal or weld heat-affected zone (HAZ) will not occur. The applicant responded that in some cases the RV will be allowed to cool to ambient temperature after application of postweld baking to remove any hydrogen that may be present. Previous BWR licensing documents, including the ABWR final SER (NUREG–1503), have included this same allowance and it has been accepted by the staff. The applicant indicated that specific postweld baking parameters are dictated by the type of weld involved, the welding process (e.g., inert gas shielded), and prior qualification testing. For example, drop of preheat is allowed for narrow gap, gas tungsten arc welding (GTAW), or gas metal arc welding (GMAW) joints when the weld is subjected to postweld baking for 2 hours at 300 degrees Celsius (C) (572 degrees Fahrenheit [F] or 4 hours at 200 degrees C (392 degrees F). With gas-shielded welding, there is little potential for the introduction of hydrogen into the weld zone in any case. In accordance with RG 1.50, all such welds will be subjected to volumetric examination to confirm the absence of delayed cracking. All such joints will subsequently receive postweld heat treatment. Therefore, the applicant contends that a combination of postweld baking and inspection meets the intent of the RG. The applicant stated that this process has been successfully applied to operating BWR RVs.

The staff considers the applicant's procedure to perform postweld baking at the temperatures and times stated above for the referenced welding processes to fabricate RCPB components to be an acceptable alternative to the guidance in RG 1.50, which provides guidance on the maintenance of preheat until postweld heat treatment is performed. In response to RAI 6.1-4, which references RAI 5.2-44, the applicant indicated that welding processes, such as flux-shielded welding, will require rigorous qualification of the effectiveness of the postweld baking.

The staff notes that this method has been successfully used in several other applications, such as fossil fuel electric generation facilities, as well as petrochemical facilities, with materials that are much more sensitive to hydrogen cracking than those materials used within the RCPB of a nuclear power plant. Postweld baking is an effective measure to prevent delayed hydrogen cracking in welds that do not go directly from preheat temperature to postweld heat treatment. The staff therefore considers the applicant's alternative to RG 1.50 acceptable, given that it provides reasonable assurance that delayed hydrogen cracking will not occur in the time that a weld is completed through completion of postweld heat treatment.

Although the staff finds the applicant's alternative to RG 1.50 acceptable, in a supplement to RAI 10.3-4, the staff requested the applicant to modify the DCD to include its alternative to RG 1.50 as it applies to all ASME Code Class 1, 2, and 3 piping and components. In addition, the staff asked that the applicant modify the DCD to include its response to RAI 6.1-4 in which it states that it will apply the minimum preheat recommendations found in ASME Code, Section III, Appendix D, Article D-1000, to all Class 1, 2, and 3 carbon steel and low-alloy steel piping and components in the ESBWR design. The staff tracked RAI 10.3-4 as an open item in the SER with open items.

The staff reviewed Revision 5 of the ESBWR DCD and verified that the applicant appropriately referenced ASME Code, Section III, Appendix D, Article D-1000, and RG 1.50 in DCD Section 5.2.3.3.2. The staff finds this reference acceptable because it meets the acceptance criteria of SRP Section 5.2.3. RAI 10.3-4 and the associated open item, as it applies to the RCPB, are resolved.

DCD Tier 2, Revision 9, Section 5.2.3.3.2, states that electroslag welding is not allowed on structural weld joints of low-alloy steel. Therefore, RG 1.34 does not apply to the ESBWR design. RG 1.43 applies to clad low-alloy steel, and the DCD identifies the RPV as the only stainless steel clad low-alloy steel component in the RCPB. Section 5.3 of this report evaluates the applicant's adherence to the guidance in RG 1.43 related to RPV fabrication.

In DCD Tier 2, Section 5.2.3.4.2, the applicant states that the ESBWR design meets the intent of RG 1.71. In RAI 5.2-45, the staff asked the applicant to discuss its deviations from specific portions of RG 1.71 and explain how those deviations meet the intent of the RG. In response, the applicant stated that restricted access qualifications are required when access to a nonvolumetrically examined production weld is less than 305 mm (about 12 inches [in]) in any direction and allows welding from one access direction only. Requalification is required if the production weld is more restricted than the welder's performance qualification. The applicant provided the following rationale:

> If a RCPB weld is subject to volumetric inspection, the inspection method and acceptance criteria will be according to ASME Section III, Subsection NB. If the

weld passes this inspection, the weld quality is considered acceptable irrespective of the access restriction. Therefore, the intent of the RG is met by inspection. The fabricator or installer must produce welds that satisfy the Code irrespective of any access restrictions.

The RG indicates restrictions of 304.8 to 355.6 mm (12 to 14 inches). Since this is insufficiently definitive from a specification and quality assurance point of view, the applicant selected 305 mm (~12 inches) as the defined limit.

Practically, even though a restriction may exist in one direction from the weld, this is not necessarily the only direction from which the welder may approach the weld. Therefore, if the welder can freely approach the weld from another direction with no access restrictions, the restricted access performance qualification is not required. It is further noted that in the ESBWR design, there are few, if any, RCPB welds that truly have restricted access. Additionally, much of the welding is performed with mechanized welding systems where physical access for a welder is not relevant to the ultimate weld quality.

The staff reviewed the applicant's alternative to RG 1.71 as stated in DCD Tier 2, Section 5.2.3.4.2. The staff has determined that the applicant's alternative is consistent with the intent of RG 1.71. The applicant's alternative will provide reasonable assurance that welders working in restricted access positions will be appropriately qualified and thus produce sound welds.

For nondestructive examination (NDE) of ferritic steel and austenitic stainless steel tubular products, compliance with the applicable provisions of the ASME Code meets the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards. Subsections NB-2550 through NB-2570, which are discussed in Section 5.2.3.3.4 of this report, are the applicable provisions of ASME Code, Section III.

5.2.3.3.4 Fabrication and Processing of Austenitic Stainless Steel

All stages of component manufacturing and reactor construction must include process control techniques, in accordance with the requirements of GDC 1, as it relates to nondestructive testing (i.e., examination) to quality standards; GDC 4; and Criterion XIII, "Handling, Storing, and Shipping," of Appendix B to 10 CFR Part 50. These requirements prevent severe sensitization of the material by minimizing exposure of stainless steel to contaminants that could lead to SCC and reduce the likelihood of component degradation or failure through contaminants.

The applicant meets the requirements of GDC 4 and Criterion XIII of Appendix B to 10 CFR Part 50 by complying with the applicable provisions of the ASME Code and following the guidance found in the regulatory positions of RG 1.31, Revision 3, "Control of Ferrite Content in Stainless Steel Weld Metal," issued April 1978; RG 1.36; RG 1.37, Revision 1, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," issued March 2007; RG 1.44; and RG 1.71.

The staff reviewed DCD Tier 2, Section 5.2.3.4, to ensure that austenitic stainless steel RCPB components are (1) compatible with environmental conditions to avoid sensitization and SCC, (2) compatible with thermal insulation, (3) have appropriate controls on welding and material preservation, and (4) receive appropriate NDE. For NDE of ferritic steel and austenitic stainless steel tubular products, the applicant complied with the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards by specifying the appropriate provisions of the ASME Code, which are in Section III, Subsections NB-2550 through NB-2570. DCD Tier 2, Revision 9, Section 5.2.3.3.3, states that seamless tubular products must be examined according to ASME Code, Section III, Subsection NB-2550, welded tubular products according to Subsection NB-2560, and cast tubular products according to Subsection NB-2570.

The DCD indicates that all austenitic stainless steels are supplied in the solution heat-treated condition, and special sensitization tests are applied to confirm and ensure proper heat treatment. In RAI 5.2-48, the staff asked the applicant to describe its "special sensitization test" that will be applied to ensure proper heat treatment. In response, the applicant indicated that the test used to detect susceptibility to intergranular attack is a modified version of ASTM A262, Practice A, wherein rejectable ditching is defined more strictly than in the ASTM version, and retest and acceptance by Practice E is not allowed. The staff considers limiting retest and acceptance by Practice E to be a conservative practice, which, therefore, meets the intent of RG 1.44.

In RAI 5.2-49, the staff asked the applicant to discuss its solution heat treatment requirements for austenitic stainless steel components and welds. In response, the applicant explained its heat treatment requirements and stated that its solution heat treatment practice is consistent with that described in RG 1.44 and NUREG–0313. The staff finds the applicant's responses to RAIs 5.2-48 and 5.2-49 acceptable because they conform with the guidance provided in RG 1.44 and NUREG–0313.

The ESBWR conforms to the guidance provided in RGs 1.31, 1.36, and 1.44.

The applicant's acceptance criteria for cleaning and cleanliness controls meet the intent of RG 1.37, Revision 1. The applicant provided an alternative to RG 1.37 that is acceptable to the staff. For a discussion refer to Section 4.5.1.2.5 of this report.

5.2.3.4 *Conclusions*

For the reasons set forth above, the staff finds that the design of the RCPB materials is acceptable and meets the requirements of GDC 1, 4, 14, 30, and 31; Appendices B and G to 10 CFR Part 50; and 10 CFR 50.55a.

5.2.4 RCS Pressure Boundary Inservice Inspection and Testing

5.2.4.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 5.2.4, in accordance with SRP Section 5.2.4, Revision 2.

GDC 32, "Inspection of reactor coolant pressure boundary," requires the periodic inspection and testing of the RCPB, and specific requirements are outlined in 10 CFR 50.55a and detailed in ASME Code, Section XI. Compliance with the preservice and inservice examinations required by 10 CFR 50.55a, as detailed in ASME Code, Section XI, partially satisfies the requirements of GDC 32, as discussed below:

- Compliance with GDC 32 requires, in part, that all components that are part of the RCPB be designed to permit periodic inspection and testing of important areas and features to assess structural and leaktight integrity. Meeting the requirements of GDC 32 ensures an effective periodic inspection program for the RCPB to identify aging effects or other incipient degradation phenomena, thus enabling licensees to take prompt preventive measures to preclude potential loss of coolant or impaired reactor core cooling.

- Compliance with 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function they are intended to perform. By reference, 10 CFR 50.55a incorporates Section XI of the ASME Code.

5.2.4.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 5.2.4, describes the PSI/ISI and system pressure test programs for NRC QG A, ASME Code Class 1 items. This section describes these programs' implementation of the requirements of Subsection IWB of ASME Code, Section XI. The design to perform PSI is based on the requirements of ASME Code, Section XI, 2001 Edition through 2003 Addenda, as specified in DCD Tier 2, Revision 9, Table 1.9-22. The applicant indicated that the COL holder is responsible for the development of the PSI/ISI program plans which must be based on the edition and addenda of ASME Code, Section XI, specified in DCD Tier 2, Revision 9, Table 1.9-22. The ASME Code requirements discussed in this section are provided for information.

5.2.4.3 *Staff Evaluation*

In DCD Tier 2, Revision 3, Section 5.2.4, the applicant stated that the development of the PSI/ISI program plans is the responsibility of the COL holder and must be based on ASME Code, Section XI, 2001 Edition through 2003 Addenda. DCD Tier 2, Section 6.6, Revision 3, indicated that the development of the ASME Code Class 2 and 3 PSI/ISI program plans would be the responsibility of the COL holder and must based on the edition and addenda of ASME Code, Section XI, specified in 10 CFR 50.55a. DCD Tier 2, Revision 3, Section 6.6, also stated that the COL holder shall specify the edition of the ASME Code to be used, based on the date of issuance of the construction permit or license, in accordance with 10 CFR 50.55a. There appeared to be an inconsistency in the DCD between the editions and addenda of ASME Code, Section XI, which COL applicants are expected to use to develop their PSI/ISI programs.

In RAI 5.2-63, the staff requested that the applicant revise DCD Tier 2, Sections 5.2.4 and 6.6, to clearly and accurately state the requirements governing the applicable ASME Code edition and addenda to be used by the COL applicant to develop PSI/ISI programs. The staff tracked RAI 5.2-63 as an open item in the SER with open items.

The applicant responded indicating that it would modify DCD Tier 2, Sections 5.2.4 and 6.6, to state that the ESBWR is designed for the performance of PSI/ISI including consideration of the requirements of the ASME Code, Section XI, edition and addenda specified in Table 1.9-22. The applicant further stated that the development of the PSI/ISI programs is the responsibility of the COL holder and will be based on the ASME Code, Section XI, edition and addenda approved in 10 CFR 50.55a(b) 12 months before initial fuel load. The staff finds this acceptable. The staff verified that the applicant made the above modifications to DCD Tier 2, Revision 4. Sections 5.2.4 and 6.6. RAI 5.2-63 and the associated open item are resolved. Subsequently, in DCD Revision 7, the applicant modified the DCD to state that the licensee will be responsible for the actual development of the PSI/ISI programs. The staff finds this acceptable because licensee is a more appropriate term than COL holder.

5.2.4.3.1 System Boundary Subject to Inspection

The applicant's definition of the RCPB is acceptable if it includes all pressure vessels, piping, pumps, and valves that are part of the RCS, or connected to the RCS, up to and including the following:

- The outermost containment isolation valve in system piping that penetrates the primary reactor containment

- The second of two valves typically closed during normal reactor operation in system piping that does not penetrate primary reactor containment

- The RCS SRVs.

The applicant stated, in DCD Tier 2, Section 5.2.4.1, that the Class 1 system boundary for both the PSI/ISI programs and the system pressure test program includes all of those items within the Class 1 and QG A boundary on the piping and instrumentation schematics. The applicant indicated that based on 10 CFR Part 50 and RG 1.26 the boundary includes the following:

- RPV
- Portions of the main steam system
- Portions of the feedwater system (FWS)
- Portions of the standby liquid control system (SLCS)
- Portions of the RWCU/SDC system
- Portions of the ICS
- Portions of the GDCS

The staff reviewed the information provided in DCD Tier 2, Section 5.2.4.1, and determined that the ASME Code Class 1 boundary requirements identified by the licensee are consistent with the acceptance criteria in SRP Section 5.2.4.II.1. Section 3.2 of this report presents a detailed staff review of the applicant's classification of ASME Code Class 1 components and piping.

5.2.4.3.2 Accessibility

The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with ASME Code, Section XI, Subsection IWA-1500, "Accessibility."

In DCD Tier 2, Section 5.2.4.2, the applicant stated that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, Subsection IWB-2500. The applicant also stated that ASME Code, Section XI, Subsection IWA-1500 defines considerations for accessibility.

DCD Tier 2, Section 5.2.4.2, states, under piping, pumps, valves, and supports, that welds are located to permit ultrasonic examination from at least one side, but where component geometries permit, access from both sides is provided. This is acceptable to the staff for ferritic welds, because a one-sided ultrasonic examination can be performed on ferritic materials. However, one-sided ultrasonic examinations cannot be performed on austenitic or DMWs using current technology. For austenitic and DMWs that are accessible from one side only, radiography would be required to attain 100-percent weld coverage for examinations required by ASME Code, Section XI. The staff is concerned that operational experience shows that radiography is not practical in some applications in current operating plants. Difficulty in draining systems and radiological concerns sometimes preclude the use of radiography, resulting in licensees requesting relief from inspection requirements. Designing a system in a manner that will require radiography must include considerations related to operating conditions and radiological concerns to ensure that ISI inspections will be practical to be performed after the plant goes into operation in order to meet the requirements of 10 CFR 50.55a(g). As discussed below, the staff issued several RAIs to address these concerns.

The staff issued several RAIs (6.6-1, 6.6-2, 6.6-3, 6.6-4, 5.2-51, 5.2-53, 5.2-54, 5.2-57, and 5.2-58) regarding the accessibility of components for inspections required by ASME Code, Section XI, and 10 CFR 50.55a. The staff developed RAI 5.2-62, which superseded the aforementioned RAIs, regarding the accessibility and inspectability of welds and components. In RAI 5.2-62, the staff requested that the applicant modify the DCD to (1) specify the inspection methods that are practical to use for ISI of welds in ASME Code Class 1 and 2 austenitic and DMWs and (2) add COL applicant items to Sections 5.2.4 and 6.6 to ensure that a COL applicant referencing the ESBWR will provide a detailed description of its plans to incorporate, during design and construction, access to piping systems to enable NDE of such welds during ISI.

ASME Code, Section XI, as incorporated into 10 CFR 50.55a(g), currently allows for either ultrasonic or radiographic examination of welds in ASME Code Class 1 and 2 piping systems. The staff asked that the applicant modify DCD Tier 1 to state that one or both of these types of examinations are practical for ISI of austenitic and DMWs. The staff notes that ultrasonic examination has advantages with respect to keeping exposures as low as reasonably achievable. With this change to the DCD, any design certification rule that might be issued for the ESBWR will preclude the granting of relief under 10 CFR 50.55a(g)(6) for ISI of such welds. The staff requested that the applicant confirm that austenitic or DMWs in Class 1 and 2 piping systems will be accessible for examination by either ultrasonic or radiographic examination, thus satisfying the requirements of 10 CFR 50.55a(g)(3).

In support of these DCD changes, a COL applicant referencing the ESBWR design certification application should inform the staff of how it plans to meet all access requirements during construction and operation, as required by 10 CFR 50.55a(g)(3)(i) and (ii). The staff notes that the PSI requirements are known at the time a component is ordered, and 10 CFR 50.55a(g) does not provide for consideration of relief requests for impractical examination during the construction phases of the component. The COL items requested above should reflect these considerations. The staff tracked RAI 5.2-62 as an open item in the SER with open items.

The applicant modified DCD, Sections 5.2.4 and 6.6 to describe its design process to ensure that the accessibility of austenitic and DMWs enable the performance of ultrasonic testing or radiographic testing. The staff reviewed the applicant’s RAI response and modifications in DCD Tier 2, Revision 5, Sections 5.2.4 and 6.6, and found them to be unacceptable because they did not address a design for accessibility which took into account operational and radiological concerns. The staff issued RAI 5.2-62 S01, and requested that the applicant address this issue.

The applicant modified DCD Tier 2, Sections 5.2.4 and 6.6 to address the staff’s concerns. Section 6.6 of this report addresses the accessibility of ASME Code Class 2 components. The applicant proposed to modify DCD Section 5.2.4.2 and include Tier 2* information in lieu of the Tier 1 changes requested by the staff. Given that the COL applicant cannot depart from Tier 2* information without NRC approval, the staff finds that the applicant’s proposed modifications described below are acceptable:

[The ESBWR design includes specific access requirements, in accordance with 10 CFR 50.55a(g)(3), to support preferred UT or optional RT examinations. The design of each component and system takes into account the NDE method, UT or RT, that will be used to fulfill PSI and inservice inspection examination and will take into full consideration the operational and radiological concerns associated with the method selected to ensure that the performance of the required examination will be practical during commercial operation of the plant. Additionally, the design procedural requirements for the 3D layout of the plant include acceptance criteria regarding access for inspection equipment and personnel]*. However, with respect to any design activities for components that are not included in the referenced ESBWR certified design, it is the responsibility of the COL applicant to preserve accessibility to piping systems to enable NDE of ASME Code Class 1 austenitic and DM welds during inservice inspection (COL Item 5.2-3-A).

The staff finds that the proposed modifications to DCD Section 5.2.4.2 and Section 5.2.6, discussed above, provide assurance that austenitic and DMWs will be accessible so that inspections required by the ASME Code may be performed, taking into account operational and radiological concerns that could affect the practicality of the inspection method chosen for PSI/ISI. The staff reviewed ESBWR DCD, Revision 6, and verified that the applicant made the modifications to the DCD discussed above. RAI 5.2-62 and its associated open item are therefore resolved.

DCD Tier 2, Section 5.2.4.2, references a formula, L=2T+152 mm, that is used in the ESBWR design to determine the minimum length (L) for spool pieces. In RAI 5.2-52, the staff asked the applicant to explain how it determined that the distance derived from the formula is adequate for the ESBWR design. In response to this RAI, the applicant stated that the basis for its minimum spool piece length of L=2T+152 mm is ASME Code, Section XI, Appendix D, which specifies L=2T+ 50.8 mm. The 50.8-mm (2.0 in.) allowance accommodates the transducer footprint. The 2T (T=thickness) distance allows for a full ultrasonic V-path for a 45-degree transducer. The additional 101.2 mm (4.0 in.) is an allowance for scanner tracks, other beam paths, and the like. The staff finds this acceptable because the applicant has considered the necessary spool piece lengths to facilitate ultrasonic testing examinations.

DCD Tier 2, Section 5.2.4.2, indicates that items such as nozzle-to-vessel welds often may have inherent access restrictions when vessel internals are installed. Therefore, preservice examination must be performed as necessary to achieve the required examination volume on these items before installation of internals, which would interfere with examination. Section 5.2.4.2 further states that access is sufficient for the inservice examination of the volume described in ASME Code Case N-613-1. The staff finds this acceptable, given that the PSI will be performed in accordance with ASME Code, Section XI, and the ISI of these components will be performed in accordance with ASME Code Case N-613-1, which the NRC endorses in RG 1.147, Revision 14. Use of NRC-endorsed ASME Code cases is permitted by 10 CFR 50.55(g)(3)(i).

The staff reviewed DCD Tier 2, Section 5.2.4.2, regarding the accessibility of RPV welds, RPV head, RPV studs, and RPV washers. The applicant has incorporated access for examinations of these components into the design of the RPV, biological shield, and vessel insulation to enable the appropriate ultrasonic and visual examinations to be conducted. This includes not only access for remotely operated ultrasonic examination devices, but also sufficient access to perform visual examination during system leakage and hydrostatic testing.

5.2.4.3.3 Examination Categories and Methods

The examination categories and methods specified in the DCD are acceptable if they are consistent with the criteria in ASME Code, Section XI, Subsection IWB-2000, "Examination and Inspection." Every area subject to examination should fall within one or more of the examination categories in Subsection IWB-2000 and must be examined, at least to the extent specified. The requirements of Subsection IWB-2000 also identify the methods of examination for the components and parts of the pressure-retaining boundary.

The applicant's examination techniques and procedures used for PSI or ISI of the system are acceptable, if they conform to the following criteria:

- The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with Subsection IWA-2000 and Subsection IWB-2000 of ASME Code, Section XI.

- The methods, procedures, and requirements regarding qualification of NDE personnel are in accordance with Subsection IWA-2300, "Qualification of Nondestructive Examination Personnel."

- The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the requirements provided in Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," to Division 1 of ASME Code, Section XI. In addition, the performance demonstration for ultrasonic examination systems reflects the requirements provided in Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of ASME Code, Section XI.

The staff reviewed DCD Tier 2, Sections 5.2.4.3.1 and 5.2.4.3.2, which discuss examination techniques, categories, and methods. The visual, surface, and volumetric examination techniques and procedures conform to the requirements of Subsection IWA-2200 and Table IWB-2500-1 of ASME Code, Section XI, and are therefore acceptable to the staff.

The ASME Code requirements discussed in Section 5.2.4 of the DCD are based on the 2001 edition of the ASME Code, Section XI, with the 2003 addenda. This edition and addenda of Section XI of the ASME Code requires the implementation of Appendix VII for qualification of NDE personnel for ultrasonic examination and the implementation of Appendix VIII for performance demonstration for ultrasonic examination of RCPB piping and components identified in Table IWB-2500. The DCD indicates that ultrasonic examination systems must be qualified in accordance with industry-accepted programs for implementation of the ASME Code, Section XI, Appendix VIII. The staff finds this acceptable.

5.2.4.3.4 Inspection Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service, hereafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of Subsection IWA-2000 concerning inspection intervals of ASME Code, Section XI.

DCD Tier 2, Section 5.2.4.4, discusses inspection intervals. Subsections IWA-2400 and IWB-2400 of ASME Code, Section XI, define inspection intervals. The inspection intervals specified for the ESBWR components are consistent with the definitions in Section XI of the ASME Code and, therefore, are acceptable.

5.2.4.3.5 Evaluation of Examination Results

The standards for evaluation of examination results are acceptable if they conform to the requirements of ASME Code, Section XI, Subsection IWB-3000, "Acceptance Standards." The proposed program for repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if it is consistent with the requirements of ASME Code, Section XI, Subsection IWA-4000, "Repair/Replacement Activities." ASME Code, Section XI, Subsection IWB-3000, describes the criteria that establish the need for repair or replacement.

DCD Tier 2, Revision 9, Section 5.2.4.5, indicates that examination results are evaluated in accordance with ASME Code, Section XI, Subsection IWB-3000, with repairs based on the requirements of Subsection IWA-4000. The staff finds this acceptable because it meets the requirements of ASME Code, Section XI.

5.2.4.3.6 System Leakage and Hydrostatic Pressure Tests

The pressure-retaining ASME Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program meets the requirements of ASME Code, Section XI, Subsection IWB-5000, "System Pressure Tests."

In DCD Tier 2, Section 5.2.4.6, the applicant described the system leakage and hydrostatic pressure test requirements. The applicant discussed those portions of ASME Code, Section XI, Subsections IWA-5000, IWB-5000, and IWB-2500, to be applied to system leakage and hydrostatic tests. The staff finds that the specific portions of ASME Code, Section XI, referenced by the applicant are acceptable. However, the staff requested that the applicant revise the DCD to clarify that all applicable requirements of Subsections IWA-5000 and IWB-5000 will apply to system leakage and hydrostatic pressure tests.

In RAI 5.2-65, the staff requested that the applicant revise DCD Tier 2, Section 5.2.4.6 and Section 6.6.6, to clarify that system leakage and hydrostatic pressure tests will meet all requirements of ASME Code, Section XI, Subsections IWA-5000, IWB-5000, IWC-5000, and IWD-5000. The staff tracked RAI 5.2-65 as an open item in the SER with open items. The applicant modified DCD, Section 5.2.4.6 to state that ASME Code Class 1 components will meet the requirements of Subsections IWA-5000 and IWB-5000. Section 6.6 of this report addresses the requirements for ASME Code Class 2 and 3 components. The staff reviewed DCD, Revision 6, and verified that the applicant had made the appropriate modifications to Section 5.2.4.6. RAI 5.2-65 and the associated open item, as they pertain to the RCPB, are resolved.

5.2.4.3.7 Augmented Inservice Inspection To Protect against Postulated Piping Failures

The augmented ISI program for high-energy fluid system piping between containment isolation valves is acceptable if the extent of ISI examinations completed during each inspection interval provides 100-percent volumetric examination of circumferential and longitudinal pipe welds with the boundary of these portions of piping. Section 6.6.3.7 of this report addresses this issue.

5.2.4.3.8 Combined License Information

DCD Tier 2, Section 5.2.6, includes COL information items pertaining to PSI/ISI and the design for accessibility.

In RAI 5.2-64, the staff asked the applicant to revise DCD Tier 2, Sections 5.2.4 and 6.6, to include a COL applicant item to provide a detailed description of the PSI/ISI programs, augmented inspection programs, and milestones for their implementation. The staff was concerned that the applicant's reference to the COL applicant, did not clearly indicate that the COL applicant must provide, in the COL application, a description of its PSI/ISI program and augmented inspection programs with commitments for their scheduled implementation. The staff understands that the COL holder will fully develop and implement the actual programs. However, the COL applicant must fully describe the PSI/ISI and augmented inspection programs to allow the staff to make a reasonable assurance finding of acceptability. The staff tracked RAI 5.2-64 as an open item in the SER with open items.

The applicant modified DCD Tier 2, Section 5.2.5 to address the staff's concerns. The staff reviewed DCD, Revision 5, and verified that the applicant had made appropriate modifications to Section 5.2.6. COL Information Item 5.2-1-A now states that the COL applicant is responsible for providing a full description of the PSI/ISI programs and augmented inspection programs, including milestones for their implementation, by supplementing, as necessary, the information in Section 5.2.4. The requirements described in DCD Tier 2, Revision 9, Sections 5.2.4.1 through 5.2.4.10 are based on ASME Code, Section XI. The staff finds this acceptable because the applicant addressed the staff concerns discussed in RAI 5.2-64 for ASME Code Class 1 systems. Section 6.6 of this report discusses RAI 5.2-64 as it relates to ASME Code Class 2 and 3 systems. RAI 5.2-64 and the associated open item, as it pertains to the RCPB, are resolved.

To address the staff's concerns, expressed in RAI 5.2-62, related to the responsibility of the COL applicant to ensure a design that provides sufficient accessibility to perform PSI/ISI, the applicant modified DCD Tier 2, Section 5.2.6 to include COL Information Item 5.2-3-A. This COL information item states that the COL applicant is responsible for developing a plan and providing a full description of its use during construction, PSI, ISI, and design activities for components that are not included in the referenced certified design to preserve accessibility to piping systems to enable NDE of ASME Code Class 1 austenitic and DMWs during ISI. The staff finds this acceptable because the COL applicant will address any design activities beyond the scope of the ESBWR design certification related to ensuring the accessibility of welds for ISI.

5.2.4.4 *Conclusions*

Based on its evaluation of the system boundary subject to inspection, accessibility, examination categories and methods, inspection intervals, evaluation of examination results, and system leakage and hydrostatic pressure tests, the staff finds that the periodic inspection and testing of the RCPB are acceptable. In addition, the inspection and test program satisfies GDC 32 because it meets the applicable requirements of ASME Code, Section XI, as endorsed in 10 CFR 50.55a.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

5.2.5.1 *Regulatory Criteria*

The staff reviewed the RCPB leakage detection system in accordance with SRP Section 5.2.5, Revision 2. Staff acceptance of the leakage detection design is based on its meeting the requirements of the following criteria:

- 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

- GDC 30, as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage

5.2.5.2 *Summary of Technical Information*

In DCD Tier 2, Revision 9, Section 5.2.5, the applicant described the RCPB leakage detection systems and the design criteria adopted to satisfy NRC regulatory requirements. The systems are designed to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage.

The following are the primary detection methods used for monitoring small unidentified leaks:

- The drywell floor drain high-conductivity waste sump pump activity
- The drywell sump level changes
- The drywell air coolers condensate flow rate
- The fission products' radioactivity

These parameters are continuously monitored and recorded in the MCR and alarmed upon abnormal indications.

The secondary methods used to detect gross unidentified leakage are the pressure and temperature parameters of the drywell atmosphere. High atmospheric pressure in the drywell trips the reactor and initiates isolation of the containment isolation valves. The ambient temperature in the drywell is also monitored and alarmed.

Identified and unidentified leakages from sources within the drywell are collected and directed to separate sumps—the drywell equipment drain low-conductivity waste sump for identified leakages and the drywell floor drain high-conductivity waste sump for unidentified leakages.

In DCD Tier 2, Revision 9, Section 5.2.6, the applicant identified COL Information Item, COL 5.2-2-A, "Leak Detection Monitoring." This COL information item requires a COL Applicant to include the following in its operating procedure development program:

- Procedures to convert different parameter indications for identified and unidentified leakage into common leak rate equivalents and leak rate rate-of-change values.

- Procedures for monitoring, recording, trending, determining the source(s) of leakage, and evaluating potential corrective action plans.

- Milestone for completing this category of operating procedures.

5.2.5.3 *Staff Evaluation*

The staff reviewed RCPB leakage detection systems for the ESBWR in accordance with SRP Section 5.2.5, Revision 2. Staff acceptance of the leakage detection design is based on whether the design meets the requirements of GDC 2 and 30. The leakage detection design conforms with GDC 2 if it meets the guidelines of RG 1.29, Revision 4, Positions C.1 and C.2. The leakage detection design conforms with GDC 30 if it meets the guidelines of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," issued May 1973, Positions C.1 through C.9.

The staff asked the applicant to provide the additional information requested in RAIs 5.2.1 through 5.2.5. The staff reviewed the applicant's responses and discusses its evaluation below.

In RAI 5.2-1(a), the staff requested the applicant to clarify the statement in DCD Tier 2, Section 5.2.5, Item (3), stating that the system is equipped with indicators and alarms for each leakage detection system in the control room and permits only "qualitative" interpretations of such indicators. In response, the applicant stated that the information presented to the MCR operators will be "quantitative," enabling the operator to convert the various readings to an equivalent leakage rate. The applicant revised the statement in DCD Tier 2 to reflect that the control room information is both qualitative and quantitative. The staff verified the revised statement in DCD Tier 2, Section 5.2.5, Revision 3. Therefore, RAI 5.2-1(a) is resolved.

In RAI 5.2-1(b), the staff asked the applicant to explain how the proposed TS and alarm limit for unidentified leakage of 18.93 liters per minute (5 gallons per minute [gpm]) is consistent with the 3.79 liters per minute (1 gpm) criterion specified in Positions C.2 and C.5 of RG 1.45. In response, the applicant maintained its position that the TS and alarm limit for unidentified leakage shall be 18.93 liters per minute (5 gpm) based on its historical BWR leakage detection and alarm limits. The applicant stated that Positions C.2 and C.5 specified only the "sensitivity" of the instrument rather than the TS or alarm limit and noted that the ESBWR instrument has the sensitivity of 3.79 liters per minute (1 gpm). RG 1.45 provides guidance on the "detector sensitivity" and states that "sumps and tanks used to collect unidentified leakage and air cooler condensate shall be instrumented to alarm for increases of from [1.89 to 3.79 liters per minute] 0.5 to 1.0 gpm." The staff found that the instrument sensitivity of 3.79 liters per minute (1 gpm) is neither specified in the TS limit nor reflected by an alarm setpoint that could provide an early warning signal to alert operators to take action. The staff considered the ESBWR alarm limit of 18.93 liters per minute (5 gpm) alone to be unacceptable because it was inconsistent with RG 1.45, as stated above, and did not serve the intended function of alerting operators to take action before the TS limit is reached. The staff tracked RAI 5.2-1 as an open item in the SER with open items.

In DCD Tier 2, Revision 6, Section 5.2.5.5, the applicant stated that a rate-of-change alarm setpoint is established at a lower limit value of 8.33 liters per minute (2.2 gpm) within one hour. The rate-of-change alarm provides an early alert for the control room operators to initiate investigation of the cause and proper response actions for the change of unidentified leakage flow before reaching or exceeding the TS limit. The staff finds that this change addresses the concern identified in RAI 5.2-1. Therefore, RAI 5.2-1 and the associated open item are resolved.

In RAI 5.2-2, the staff asked why ESBWR TS Limiting Condition for Operation 3.4.2 specified a more relaxed limit of 18.93 liters per minute (5 gpm) for the unidentified RCPB leakage than the limit of 3.79 liters per minute (1 gpm) specified for the ABWR and for all other advanced reactors. The more relaxed limit could lead to higher operating RCPB leakage rates, fewer RCPB leakage controls, a potentially more humid environment inside containment, and an increased probability of material degradation from corrosion. In response to RAI 5.2-2, the applicant stated that an evaluation of the effects of relative humidity, including that attributable to the proposed leakage limit of 18.93 liters per minute (5 gpm), would be part of the equipment qualification requirements in the procurement of equipment. In addition, the applicant stated that the design of the ESBWR has been improved to reduce the likelihood of leaks resulting from SCC, and historically, good operator practice plays a role in the event of an anomaly in unidentified leakage. Typical operator practice will investigate, record, track, and evaluate trends in leakage and take necessary measures to locate, assess, and repair the source of any leakage. The staff agreed that the material design improvement can reduce the likelihood of leaks resulting from SCC, but the improvement cannot eliminate all possible leaks. The staff also agreed that good operator actions at low-level leakage below the TS limit are acceptable measures to address the concern of long-term leakage. To account for the good operator practice, every COL applicant should have operating procedures to manage low-level RCS leakage, and the alarm limit shall be set as low as practicable to provide an early warning signal to the operators to implement the procedures. As a result of discussions between the applicant and the staff, the applicant agreed to add a COL applicant item in DCD Tier 2, Revision 3, Section 5.2.6. This item stated that "operators will be provided with procedures to assist in monitoring, recording, trending, determining the source of leakage, and evaluating potential corrective action." The staff found the statement unacceptable because it did not indicate that the procedures are for low-level leakage (lower than the TS limit) and did not indicate that the COL holder is responsible for the development of the procedures. In addition, the design needs an appropriate alarm limit (resolution of open item associated with RAI 5.2-1) to provide an early warning signal to the operators to implement the procedures. The staff tracked RAI 5.2-2 as an open item in the SER with open items.

The staff issued supplemental RAI 5.2-2 S03 and RAI 5.2-1 S03 requesting the applicant to address open items associated with RAI 5.2-2. In response to these supplemental RAIs, the applicant revised DCD Tier 2, Section 5.2.6, COL Information Item 5.2-2-A, for the low-level leakage alarm set point and the operating procedure for responding to prolonged, low-level reactor coolant leakage. Further, this COL information item is described in more detail in DCD Tier 2, Revision 9, Section 5.2.5.9. The applicant stated that the COL licensee is responsible for the development of procedures for monitoring, recording, trending, determining the sources of leakage, and evaluating potential corrective action plans. In addition, in DCD Tier 2, Revision 7, Section 5.2.5.9, the applicant stated that an unidentified leakage rate-of-change alarm provided operators an early alert to initiate response actions before reaching the TS limit. The staff finds that the above changes in Revision 7 of DCD Tier 2 satisfactorily address the concern identified in RAI 5.2-2. Therefore, RAI 5.2-2 and associated open item are resolved.

In RAI 5.2-3, the staff asked the applicant to explain why the TS basis, TS B.3.4.2, "RCS Operational Leakage," refers to GDC 55, "Reactor coolant boundary penetrating containment," but not to GDC 30 as the bases for the TS. GDC 55 discusses the requirements for containment isolation valves, and GDC 30 specifies the quality of the RCPB. In response, the applicant indicated that it referenced GDC 55 in the context of defining the RCS pressure boundary and referenced GDC 30 in the bases for TS 3.3.4.1, "RCPB Leakage Detection Instrumentation." The staff reviewed the response and found that, although referencing GDC 55 is acceptable in the context of ESBWR TS B.3.4.2, it was not acceptable without also referencing GDC 30 in TS B.3.4.2. GDC 55 does not require any limit for operational leakage and does not provide any bases for requiring leakage limits, as specified in TS 3.4.2. RG 1.45 provides the guidance for implementing the requirements of GDC 30, and RG 1.45, Position C.9, states that the TS shall include the limiting conditions for identified and unidentified leakage and address the availability of various types of instruments to ensure adequate coverage at all times. This regulatory position, which implements the requirements of GDC 30, provides the bases for both TS 3.3.4.1 and TS 3.4.2. In a conference call held on August 14, 2006, the applicant agreed to revise TS B.3.4.2 by referencing GDC 30 in the bases for TS 3.4.2. The staff confirmed this change in Revision 3 of DCD Tier 2. Therefore, RAI 5.2-3 is resolved.

In RAI 5.2-4, as it relates to RG 1.45, Position C.7, the staff asked the applicant to clarify whether the procedures that will provide operator guidance on converting leakage instrument indications into a common leakage rate equivalent were generic for the ESBWR design or were to be developed by COL applicants. In response, the applicant stated that COL applicants would provide the procedures to convert different sources of leakage into a common rate equivalent. This COL item should be added to DCD Tier 2, Section 5.2.6. Accordingly, the applicant provided a markup page for Section 5.2.6 in the RAI response. However, when reviewing DCD Tier 2, Revision 2, the staff could not find the promised COL item. In a conference call on January 16, 2007, the applicant agreed to incorporate the change in Revision 3 of DCD Tier 2. In its review of Revision 3, the staff found that Section 5.2.6 stated that “operators will be provided with a procedure to determine the identified and unidentified leakage in order to establish whether the leakage rates are within the allowable TS.” The staff found this statement unacceptable for two reasons. First, the statement should identify the COL holder as responsible for the development of the procedures, and second, the statement shall better characterize the purpose of the procedures. The purpose is to convert different sources of leakage (such as sump pump activity, sump level, condensate flow rate, and radioactivity) into a common rate equivalent (expressed in gpm). Operators can use this leak rate information to monitor the leakage and to keep the leakage well below the TS limit. The purpose of the procedures is not limited to establishing whether the leakage rates are within the allowable TS. RAI 5.2-4 was being tracked as confirmatory item in the SER with open items.

In Revision 7 of DCD Tier 2, COL Information Item 5.2-2-A, and Section 5.2.5.9, the applicant stated that the Licensee is responsible for the development of a procedure to convert different parameter indications for identified and unidentified leakage into common leak rate equivalents and leak rate rate-of-change values. In DCD Tier 2, Section 5.2.5.9, the applicant stated that typical monitoring includes parameters such as sump pump run time, sump level, condensate transfer rate, and process chemistry/radioactivity. Furthermore, the monitored leakage equivalents provide information used by the plant operators to manage the leakage, establish whether the leakage rates are within the allowable TS, and determine the trend. The staff finds that the changes in Revision 7 of DCD Tier 2 address the staff concern identified above. Therefore, RAI 5.2-4 and associated confirmatory item are resolved.

In RAI 5.2-5, as it relates to the capability of the leak detection instrument to maintain and perform its safety functions following an earthquake, the staff asked the applicant to clarify which of the leak detection instrumentation discussed in DCD Tier 2, Section 5.2.5.2, is required to perform the containment isolation function and which is not. The leak detection instrumentation required to perform the isolation function is classified as Class 1E, seismic Category I, and therefore, should be consistent with the guidelines of RG 1.29. In response, the applicant revised DCD Tier 2, Section 5.2.5, to identify the leak detection instruments that are used for isolation functions and the instruments that are not used for isolation functions. The staff confirmed that this modification appeared in DCD Tier 2, Revision 6. In addition, the applicant stated that, with one exception, leak detection instruments that are not required for isolation functions are not required to remain functional following an earthquake. The exception is the drywell fission product radiation monitoring system, which is seismically qualified and should be designed in a manner that is consistent with the guidance of Positions C.1 and C.2 of RG 1.29. The staff finds the applicant's response acceptable and determined that the ESBWR design satisfies Position C.6 of RG 1.45 and Positions C.1 and C.2 of RG 1.29. Therefore, the design satisfies GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake. RAI 5.2-5 is resolved.

The staff reviewed DCD Tier 2, Chapter 16, relating to the TS of the RCPB leakage detection and issued RAI 16.2-1 (gaseous radiation monitor) and RAI 16.2-4 (rate-of-change limit in RCS operational leakage). Chapter 16 of this report discusses the responses to, and resolution of, these RAIs.

5.2.5.4 *Conclusions*

Based on the above, the staff finds that the applicant has met the requirements of GDC 2 with respect to the systems' capability to maintain and perform their safety functions in the event of an earthquake by meeting Positions C.1 and C.2 of RG 1.29 and the requirements of GDC 30, as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The staff reviewed DCD Tier 2, Revision 9, Section 5.3.1in accordance with SRP Section 5.3.1, Revision 2. The applicant's RV materials are acceptable if they meet codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure that the materials meet the relevant requirements of 10 CFR 50.55a; Appendix G and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50; and GDC 1, 4, 14, 30, 31, and 32. These requirements are discussed below:

- GDC 1 and 30 and 10 CFR 50.55a(a)(1) require SSCs important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 4 requires SSCs important to safety to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 14 requires the RCPB to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31 requires the RCPB to be designed with sufficient margins to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and minimize the probability of rapidly propagating fracture.

- GDC 32 requires the RCPB components to be designed to permit an appropriate material surveillance program for the RV.

- Appendix G to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G. In accordance with Appendix G, the RV beltline materials must have Charpy upper-shelf energy (USE) values, in the transverse direction for base material and along the weld for weld material, of no less than 102 Newton-meters (N-m) (75 foot-pound [ft-lb]) initially and must maintain Charpy USE values throughout the life of the vessel of no less than 67.8 N-m (50 ft-lb).

- Appendix H to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in the fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance to ASTM E185, "Compliance with Appendix H" satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine whether they meet the relevant requirements of Appendix H for determining and monitoring material fracture toughness.

5.3.1.1 *Summary of Technical Information*

5.3.1.1.1 Material Specifications

The applicant indicated that the material specifications are consistent with ASME Code requirements. All ferritic RV materials must comply with the fracture toughness requirements of 10 CFR 50.55a and Appendices G and H to 10 CFR Part 50.

DCD Tier 2, Revision 9, Table 5.2-4, identifies the materials used in the RV and appurtenances. The chemical compositions of the ferritic materials of the RV beltline are restricted to the maximum limits shown in DCD Tier 2, Revision 9, Table 5.3-1. Copper, nickel, and phosphorus content is restricted to reduce sensitivity to irradiation embrittlement in service.

5.3.1.1.2 Special Processes Used for Manufacturing and Fabrication

The RV is constructed primarily from low-alloy, high-strength steel plate and forgings. Plates are ordered to ASME Code SA-533, Type B, Class 1, and forgings to ASME Code SA-508, Grade 3, Class 1, specifications. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low-alloy steels. The shells and vessel heads are made from formed plates or forgings, whereas flanges and nozzles are made from forgings. Welding performed to join these vessel components is consistent with procedures qualified in accordance with the requirements of Sections III and IX of the ASME Code. GTAW, GMAW, shielded metal arc welding, and submerged arc welding (SAW) processes may be employed. Electroslag welding is not used except for cladding.

Postweld heat treatment of all low-alloy welds is performed in accordance with ASME Code, Subsection NB-4620 (DCD Tier 2, Revision 9, Table 5.3-1). The materials, fabrication procedures, and testing methods used in the construction of the ESBWR RV meet or exceed the requirements of ASME Code, Section III, Class 1 vessels.

The RV assembly components are classified as ASME Code Class 1. Complete stress reports on these components are prepared in accordance with ASME Code requirements. NUREG–

0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” issued November 1980, is also considered for FW nozzle and other such RV inlet nozzle designs. Action Plan Item A-10, “BWR Feedwater Nozzle Cracking,” is considered resolved through compliance with NUREG–0619, consistent with the NRC resolution, and compliance with GL 81-11, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (NUREG–0619).”

The staff’s review of DCD Tier 2, Section 5.3.1, in accordance with SRP Section 5.3.1, identified areas in which additional information was necessary to complete the review of the RV materials.

In response to RAI 5.3-1, the applicant confirmed that the carbon content is limited so as not to exceed 0.02 percent in all welded wrought austenitic stainless steel components in the ESBWR that are exposed to reactor water at temperatures exceeding 93 degrees C (200 degrees F). The applicant also stated that in Table 5.2-4 strength is the only distinction between 304 and 304L/316 and 316L. The applicant updated DCD Tier 2, Revision 2, Table 5.2-4 indicating that for these components the maximum allowable carbon content is 0.02 percent. The applicant’s response is acceptable to the staff. RAI 5.3-1 is closed.

In response to RAI 5.3-2, the applicant confirmed that the current practice for welding stub tubes to the bottom head is automatic GTAW. The inclusion of manual welding in the DCD is to allow for local repair using manual GTAW or GMAW. The applicant also confirmed that all weld metal is Alloy 82 with stabilization parameter control. Use of Alloy 182 is prohibited in components that come into contact with reactor water. The applicant modified the appropriate DCD sections accordingly. The applicant’s response and revisions to the DCD are acceptable to the staff. RAI 5.3-2 is closed.

In response to RAI 5.3-3, the applicant stated the following:

- Several BWR RVs have been site assembled. This includes Vermont Yankee, Monticello, Leibstadt, Clinton, and Limerick. The process for ESBWR has not been finalized at this time, but it is anticipated that the nearly completed RV will be shipped to the site in two or possibly three sections. Joining of the sections at the site may be done with the vessel axis vertical using mechanized welding equipment. Alternately, temporary rollers may be set up at the site, and the closure weld completed with mechanized SAW or GMAW.

- Local post-weld heat treatment, as allowed by ASME Code, Section III, will be performed on the circumferential weld(s). This is a relatively simple operation because the weld joins two axisymmetric cylinders of uniform thickness. The goal is to locate the welds away from discontinuities. Finite-element analysis will be used to establish the heating pattern and define temperature gradients away from the heated band. This will be followed by stress analysis to demonstrate that stresses in the adjacent material are maintained at acceptable levels. This approach has previously been successfully used to apply local post-weld heat treatment to RV nozzles where reapplication of nozzle butters was required. Likewise, this approach is routinely used to attach main steam nozzle extension forgings of low alloy steel to the steam nozzle at the ABWR construction sites. Local heat treatment of the final closing weld has been standard practice by some European manufacturers (e.g., the Cofrentes RPV) since most of their furnaces do not have the capacity to heat treat a complete RPV. The local heat treatments were performed using either heating pads or induction heating.

In response to RAI 5.3-3, the applicant also confirmed that the process of assembling the RV at a plant site has not yet been finalized. Thus, the staff finds that the fabrication process and examination process will be verified using the ITAAC described in DCD Tier 1, Revision 9, Table 2.1.1-2, Item 5.

5.3.1.1.3 Special Methods for Nondestructive Examination

The NDE of the RV and its appurtenances is conducted in accordance with the requirements of ASME Code, Section III. Volumetric examination and surface examination are performed on all pressure-retaining welds, as required by ASME Code, Section III, Subsection NB-5320. In addition, all pressure-retaining welds are given a supplemental ultrasonic preservice examination in accordance with ASME Code, Section XI. The ultrasonic examination method, including calibration, instrumentation, scanning, and coverage, is based on the requirements of ASME Code, Section XI, Appendix I.

5.3.1.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

DCD Tier 2, Revision 9, Section 5.3.1.4, addresses issues raised in the following RGs affecting the RV:

- RG 1.31 addresses controls on stainless steel welding, which are discussed in DCD Tier 2, Revision 9, Section 5.2.3.4.2.

- RG 1.34 is not applicable to the ESBWR vessel because electroslag welding is not used in structural low-alloy steel welds.

- RG 1.37 provides quality assurance guidance for the cleaning of systems and components on the site during and at the completion of construction. This cleaning follows written procedures that provide for cleanliness and ensure that the components are not exposed to materials or practices that may degrade their performance. For components containing stainless steel, RG 1.37 presents the procedures. The procedures prohibit contact with low-melting-point compounds and substances that are known to cause SCC or that can release, in any manner, substances that can cause such problems. In addition, controls are placed on the use of grinding wheels and wire brushes, which ensures that they cannot introduce degrading materials either through prior usage or through their materials of construction. In this context, degradation includes SCC. Controls also limit the introduction of unnecessary dirt and require restrictions on dirt-producing processes, such as welding or grinding, which include prompt cleaning.

- RG 1.43 is not applicable to the ESBWR vessel because the RV is constructed from low-alloy steel forgings or plates conforming to the SA-508, Grade 3, or SA-533, Type B, specification which are produced to fine grain practice. Therefore, underclad cracking is not a concern.

- RG 1.44 addresses the control of sensitization of stainless steel by the use of service-proven low-carbon materials and appropriate design and processing steps, including solution heat treatment, control of welding heat input, control of heat treatment during fabrication, and control of stresses.

- RG 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Code, Sections III and IX. The preheat temperature employed for welding of low-alloy steel meets or exceeds the recommendations of ASME Code, Section III, Appendix D. Components are either held for an extended time at preheat temperature to ensure removal of hydrogen or preheat is maintained until postweld heat treatment.

- RG 1.71 addresses welder qualification for areas of limited accessibility, which is addressed in DCD Tier 2, Revision 9, Section 5.2.3.4.2.

- RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” issued May 1988, addresses ways to predict changes in transition temperature and USE.

The staff finds that the applicant’s use of the RGs, mentioned above, to ensure the integrity of the RV is acceptable.

5.3.1.1.5 Fracture Toughness

In DCD Tier 2, Section 5.3.1.5, the applicant described the methods, codes, and standards used to comply with the requirements for fracture toughness testing in Appendix G to 10 CFR Part 50. Specifically the applicant addressed the material test coupons, location and orientation of test specimens, records and procedures for impact testing, Charpy curves for the RPV beltline, bolting material, and fracture toughness margins to control reactivity. The staff’s evaluation is provided in Section 5.3.1.2 of this report.

5.3.1.1.6 Material Surveillance

Appendix H to 10 CFR Part 50 presents the requirements for a material surveillance program for operating reactors. The purpose of the material surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region, which result from exposure of these materials to neutron irradiation. Material surveillance is accomplished using surveillance capsules, which are holders of archival beltline material and fast neutron (i.e., neutrons with energy greater than 1.0 million electron volts (MeV) dosimeters. Assessment of the irradiated material samples yields a measure of the embrittlement, and measurement of the dosimeter activation estimates the irradiation exposure.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” issued March 2001, which is based on GDC 14, 30, and 31, describes methods and practices acceptable to the staff regarding calculational techniques and statistical practices using the dosimetry measurements. In addition, the results of the dosimetry are used to benchmark and validate calculational methods for estimating vessel irradiation.

DCD Section 5.3.1.6.1 states that RV material surveillance specimens are provided in accordance with the requirements of ASTM E185 and Appendix H to 10 CFR Part 50. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a forging actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld HAZ material. The base metal and weld are heat treated in a manner that simulates the actual heat treatment performed on the beltline region of the completed vessel. Each in-reactor surveillance capsule contains 36 Charpy V-notch and 6 tensile specimens. The capsule loading consists of 12 Charpy V-notch specimens each of base metal, weld metal, and HAZ material and 3 tensile specimens each from base metal and weld metal. A set of out-of-reactor beltline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors are located within the capsules, as required by ASTM E185.

Four capsules are provided to monitor the 60-year design life of the vessel. This exceeds the three capsules specified in ASTM E185, as required by Appendix H to 10 CFR Part 50, since the predicted transition temperature shift is less than 55.6 degrees C (100 degrees F) at the inside surface of the low-alloy steel vessel.

The following proposed withdrawal schedule is modified from the ASTM E-185 schedule to monitor the RV for its 60-year design life:

- First capsule: After 6 effective full-power years (EFPY)

- Second capsule: After 20 EFPYs

- Third capsule: With an exposure not to exceed the peak end of life (EOL) fluence

- Fourth capsule: Schedule to be determined based on results of first three capsules, in accordance with ASTM E185, paragraph 7.6.2

In response to RAI 5.3-4, the applicant explained that achieving a lead factor exceeding 1.0 is relatively easy in the ESBWR because there are no obstructions in the annulus that restrict placement of the capsule holders. The location of the axial and circumferential flux peaks are known from fluence calculations, and the capsule holders can be placed precisely at these peak locations (there are a total of eight peak locations). Since the capsule holder is mounted somewhat inboard of the vessel wall, a lead factor greater than 1.0 is assured. The applicant modified DCD Tier 2, Section 5.3.4, Revision 3, and confirmed that the COL applicant will identify the following information (see COL Information Item 5.3-2-A):

- Specific materials in each surveillance capsule
- Capsule lead factors
- Withdrawal schedule for each surveillance capsule
- Neutron fluence to be received by each capsule at the time of its withdrawal
- Vessel EOL peak neutron fluence

In response to RAI 5.3-5, the applicant stated that, like all BWRs, the ESBWR will operate at a nominal temperature of about 288 degrees C (550 degrees F). However, in DCD Tier 2, Revision 9, Section 5.3.1.6, the applicant included a statement that since the vessel beltline may be exposed to a coolant temperature of minimum 271 degrees C (520 degrees F) during full power operation, the influence of the additional shift in the temperature between 288 degrees C (550 degrees F) and 271 degrees C (520 degrees F) will be added in the pressure-temperature (P/T)-curve calculation. The effect of temperatures less than 274 degrees C (525 degrees F) on irradiation embrittlement will be accounted for.

The applicant's schedule for removing the capsules for postirradiation testing includes the withdrawal of four capsules, in accordance with ASTM E185-82 (i.e., the 1982 edition of

ASTM E185) and Appendix H to 10 CFR Part 50. Staff's evaluation is provided in Section 5.3.1.2 of this report.

5.3.1.1.7 Reactor Vessel Fasteners

As described in DCD Tier 2, Revision 9, Table 5.3-1, the materials for the fasteners for the RV are controlled as follows:

- Closure studs, nuts, and washers for the main closure flange are composed of ASME Code SA-540, Grade B23 or Grade B24 material with a minimum yield strength level of 893 MPa (129.5 kilo pound/square inch [ksi]).

- Maximum measured ultimate tensile strength of the stud bolting materials must not exceed 1172 MPa (170 ksi).

5.3.1.2 *Staff Evaluation*

The staff reviewed DCD Tier 2, Revision 9, Section 5.3.1, in accordance with SRP Section 5.3.1.

The staff also reviewed the ESBWR RV materials to ensure that they meet the relevant requirements of GDC 1 and 30 and 10 CFR 50.55a(a)(1), as they relate to material specifications, fabrication, and NDE. Compliance with these requirements will determine whether the RV materials are adequate to ensure a quality product commensurate with the importance of the safety function to be performed. The material specifications for the ESBWR design are consistent with the requirements of ASME Code, Section III, and Appendix G to 10 CFR Part 50. In addition, the design and fabrication of the RV conforms to the requirements of ASME Code, Section III, Class 1. Furthermore, the RV and its appurtenances are fabricated and installed in accordance with ASME Code, Section III, Subsection NB-4100. The NDE of the RV and its appurtenances is conducted in accordance with ASME Code, Section III, requirements. Examination of the RV and its appurtenances by NDE complies with ASME Code, Section III, Subsection NB-5000. The applicant stated that all plates, forgings, and bolting are 100-percent ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods, as required by ASME Code, Division 1, Section III.

The staff finds this acceptable because compliance with the requirements of ASME Code, Section III, and Appendix G to 10 CFR Part 50 constitutes an adequate basis for satisfying the requirements of GDC 1 and 30 and 10 CFR 50.55a(a)(1) as they relate to the material specifications, fabrication, and NDE of RV materials.

Section 5.2.3 of this report provides the staff's evaluation of the welding of ferritic steels and austenitic stainless steels and addresses GDC 4.

DCD Tier 2, Revision 9, Table 5.3-1, provides the maximum limits for the elements in the materials of the RV beltline. Specified limits for RV materials used in the core beltline region are the following:

- Base Materials: 0.05-percent maximum copper, 0.006-percent maximum phosphorus, 1.0-percent maximum nickel (forging), and 0.73-percent maximum nickel (plate)

- Weld Materials: 0.05-percent maximum copper, 0.008-percent maximum phosphorus, 1.0-percent nickel, and 0.05-percent maximum vanadium

Table 5.3-1 also provides the maximum limits for the RV studs, nuts, and washers for the main closure flange.

The tests for fracture toughness of RV materials specified in the DCD are consistent with ASME Code, Section III, Subsection NB-2300, and Appendix G to 10 CFR Part 50. The staff confirmed that the applicant's initial Charpy V-notch minimum upper-shelf fracture energy levels for the RV beltline base metal transverse direction and welds are 101.7 N-m (75 ft-lb). DCD Tier 2, Revision 9, Table 5.3-3, indicates that the EOL values for the USE are greater than 67.8 N-m (50 ft-lb) for the beltline forgings and welds. The staff confirmed this by using the calculations of RG 1.99 for the beltline forgings and welds. The predicted EOL Charpy USE and adjusted reference temperature (ART) for the RV materials comply with the requirements of Appendix G to 10 CFR Part 50. The fracture toughness tests required by the ASME Code and Appendix G provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RV. This methodology will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G to 10 CFR Part 50 satisfies the requirements of GDC 14 and 31 and 10 CFR 50.55a regarding the prevention of fracture of the RV. Therefore, the staff finds that the applicant has adequately met the requirements of GDC 14 and 31 and 10 CFR 50.55a for the RV.

The design of a RV must consider the potential embrittlement of RV materials as a consequence of neutron irradiation and the thermal environment. GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV. Appendix H to 10 CFR Part 50 details the requirements of such a program.

The applicant explained that, since there are no obstructions in the annulus that restrict placement of the capsule holders, achieving a lead factor of greater than 1.0 is relatively easy in the ESBWR. The location of the axial and circumferential flux peaks are known from fluence calculations, and the capsule holders can be placed precisely at these peak locations (there is a total of eight peak locations). Mounting the capsule holder somewhat inboard of the vessel wall ensures a lead factor greater than 1.0. The applicant also confirmed that it will perform an analysis defining the lead factors and the azimuth locations of the surveillance holders. DCD Tier 2, Section 5.3.4 includes this as a COL Information Item 5.3-2-A. To meet the requirements of GDC 32, the ESBWR design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the RV beltline materials to neutron radiation. Appendix H to 10 CFR Part 50 requires that the surveillance program for the ESBWR RV meets the recommendations of ASTM E185. ASTM E185 applies to plants designed for a 40-year life, whereas the design life of the ESBWR is 60 years. ASTM E185 recommends a minimum of three surveillance capsules for an RV with an EOL shift of less than 38 degrees C (100 degrees F). The ESBWR surveillance capsule program includes four specimen capsules, with archive materials available for additional replacement capsules. The staff verified that the surveillance test materials will be prepared from samples taken from the materials used in fabricating the beltline of the RV. In addition, the staff verified that the base metal, weld metal, and HAZ materials included in the program will be those predicted to be most limiting in terms of setting pressure-temperature (P/T) limits for operation of the reactor to compensate for radiation effects during its lifetime. The staff finds that the materials selection, withdrawal, and testing requirements for the ESBWR design are consistent with those recommended in ASTM E185-82. Compliance with the materials surveillance requirements of Appendix H to 10 CFR Part 50 and ASTM E185 satisfies the requirements of GDC 32 for an appropriate surveillance program for the RV. Thus, the ESBWR design meets the requirements of GDC 32.

The applicant indicated that the material used to fabricate the closure studs will meet the fracture toughness requirements of Section III of the ASME Code and Appendix G to 10 CFR Part 50. NDE of the studs will be performed according to Section III of the ASME Code, Subsection NB-2580. In addition, ISI will be performed according to Section XI of the ASME Code, supplemented by Subsection NB-2545 or NB-2546. Conformance with the recommendations of RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” issued October 1973, ensures the integrity of the ESBWR RV closure studs and satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a. Compliance with the recommendations of RG 1.65 also satisfies the requirement in GDC 31 for the prevention of fracture of the RCPB and the requirements of Appendix G to 10 CFR Part 50, as detailed in the provisions of Section III of the ASME Code.

Generic Letter GL 92-01

GL 92-01, “Reactor Vessel Structural Integrity,” addressed NRC concerns regarding compliance with the requirements of Appendices G and H to 10 CFR Part 50, which address fracture toughness requirements and reactor vessel materials surveillance program (RVMSP) requirements, respectively. Specifically, NRC had concerns about Charpy upper shelf energy predictions for end of life for the limiting beltline weld and the plate or forging, RVs constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, and use of RG 1.99, Revision 2, to estimate the embrittlement of the materials in the RV beltline. In addition, the NRC was concerned about RVMSP compliance with ASTM E185, which requires that the licensee take sample specimens from actual material used in fabricating the beltline of the RV.

The ESBWR DCD, Revision 5, stated that the RV materials surveillance specimens are provided in accordance with the requirements of ASTM E185. Section 5.3.1.6.1 stated that the materials for the program are selected to represent materials used in the reactor beltline region and that the specimens are manufactured from forgings actually used in the beltline region and a weld typical of those in the beltline region, thus representing base metal, weld material and the weld HAZ material. Therefore, the applicant has addressed the entire beltline region in their RVMSP. The DCD Tier 2, Revision 9, also states that the predictions for changes in transition temperature and upper shelf energy are made in accordance with the guidance of RG 1.99

Finally, COL applicants referencing the ESBWR DCD are required to develop a description of their RVMSP that will include (1) specific materials in each surveillance capsule; (2) capsule lead factors; (3) withdrawal schedule for each surveillance capsule; (4) neutron fluence to be received by each capsule at the time of its withdrawal; and, (5) vessel end-of-life peak neutron fluence (This is identified as COL Information Item 5.3.2-A in DCD Tier 2, Revision 9, Section 5.3.4)

The staff finds that the applicant has met the intent of GL 92-01. In addition, a COL applicant that incorporates by reference the ESBWR DCD and provides an acceptable response to the COL items should also meet the intent of the GL. Furthermore, a COL applicant will continue to meet the intent of the GL in the future by providing the summary test reports, in accordance to ASTM E185-82, to the NRC upon withdrawal of each surveillance capsule.

Task Action Plan Item A-10

As discussed in NUREG–0933, Task Action Plan Item A-10, "BWR feedwater Nozzle Cracking," addresses the issue of cracks found during the inspection of the FW nozzles of 20 RVs. It was determined that cracking was due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low FW temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarge from high pressure and thermal cycling associated with startups and shutdowns.

ESBWR DCD, Revision 5, Section 3.9.3.2 states that RPV assembly components are classified as ASME Class 1. Complete stress reports on these components are prepared in accordance with the Code requirements. The guidance from NUREG–0619 and associated GLs 80-95 and 81-11 is factored into the FW nozzle and sparger design. The FW nozzle/sparger design does not allow incoming FW flow to have direct contact with the nozzle bore region, and the double thermal sleeve design adds further protection against thermal cycling on the nozzle. Task Action Plan Item A-10 is considered resolved through compliance with NUREG–0619. In DCD Table 1.11-1, the applicant has proposed to resolve the Task Action Item through compliance with NUREG–0619. Therefore, the staff finds the applicant has appropriately addressed this issue.

Task Action Plan Item A-11

As discussed in NUREG–0933, Task Action Plan Item A-11 addresses the issue that, because of the remote possibility that nuclear RPVs designed to the ASME Code might fail, the design of nuclear facilities must provide protection against RV failure.

Prevention of RV failure depends primarily on maintaining the RV material fracture toughness at levels that will resist brittle fracture during plant operation. As plants accumulate more service time, neutron irradiation reduces the material fracture toughness and initial safety margins. This issue is considered resolved through compliance with NUREG–0744, "Resolution of the Task A-11, Reactor Vessel Materials Toughness Safety Issue," dated October 1982, and GL 82-26, "NUREG–0744, Revision 1, Pressure Vessel Material Fracture Toughness," dated November 12, 1981. This issue did not result in establishing new regulatory requirements.

DCD Tier 2, Revision 9, Section 5.3.1.1 states that the ESBWR RV design complies with the provisions of ASME Section III, and should also meet the requirements of ASME Code Section II materials and Appendix G to10 CFR Part 50. The fracture toughness tests required by these regulations provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. In addition, DCD Tier 2, Revision 9, Table 1.11-1 the applicant proposed to resolve Task Action Plan Item A-11 through compliance with NUREG–0744. This approach is acceptable to the staff and therefore the applicant has appropriately addressed this issue.

Issue 111: Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments

As discussed in NUREG–0933, Issue 111 addresses stress corrosion cracking of ferritic steels. The cracks, first detected on the shell side of steam generator #32 of Indian Point Unit 3 (IP-3), were found to be caused by a low cycle corrosion fatigue phenomenon with cracks initiating at areas of localized corrosion and propagating by fatigue. The cause of the pitting/cracking was considered to be related to high dissolved oxygen levels and copper species in solution. Further testing indicated that the water chemistry control at IP-3 has been poor for some time.

Investigation of recent history current BWR vessels and the proposed design of ESBWR reveal that no stress corrosion cracking was ever observed in low alloy steel. The ESBWR vessel is clad with stainless steel or Ni-Cr-Fe alloy and will go through ASME Section XI inspection (see Section 5.3.3.3 of this report). Also, there will be no copper tubing in the ESBWR heat exchangers and therefore there will be no copper species in the reactor water solution as was found in the IP-3 steam generator. Finally, the ESBWR reactor water cleaning/shutdown cooling system will measure conductivity, dissolved oxygen, pH, chloride, silica, etc. as part of the sampling program guidance described in SRP Section 9.3.2 (ESBWR DCD Tier 2, Revision 9, Table 9.3-1).

Therefore, the staff finds that Issue 111 is not applicable to the ESBWR vessel.

5.3.1.3 *Conclusions*

The staff finds that the ESBWR RV material specifications, RV manufacturing and fabrication processes, NDE methods of the RV and its appurtenances, fracture toughness testing, material surveillance, and RV fasteners are acceptable and meet the material testing and monitoring requirements of Section III of the ASME Code; Appendices G and H to 10 CFR Part 50; and 10 CFR 50.55a, which provide an acceptable basis for satisfying the requirements of GDC 1, 14, 30, 31, and 32.

5.3.2 Pressure-Temperature Limits

The staff reviewed DCD Tier 2, Revision 9, Section 5.3.2, in accordance with SRP Section 5.3.2, Revision 2. The applicant's P/T limit curves are acceptable if they meet codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure that the applicant meets the relevant requirements of 10 CFR 50.55a; Appendix G to 10 CFR Part 50; and GDC 1, 14, 31, and 32. These requirements are discussed below:

- GDC 1 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31 requires that the RCPB be designed with sufficient margin to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and minimize the probability of rapidly propagating fracture.

- GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV.

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the RCPB in nuclear power plants. The staff evaluates the P/T limit curves based on Appendix G to 10 CFR Part 50, RG 1.99, and SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 requires that P/T limit curves for the RV be at least as conservative as those obtained by applying the methodology of ASME Code, Section XI, Appendix G.

RG 1.99 contains methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation. SRP Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RV based on the linear elastic fracture mechanics methodology of ASME Code, Section XI, Appendix G. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration ASME Code, Section XI, Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions; for hydrostatic testing limits, Appendix G to the ASME Code requires a safety factor of 1.5.

The methods of Appendix G to the ASME Code postulate the existence of a sharp surface flaw in the RV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-quarter of the RV beltline thickness and a length equal to 1.5 times the RV beltline thickness. The critical locations in the RV beltline region for calculating heatup and cooldown P/T curves are the one-quarter thickness (1/4T) and three-quarter thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw if initiated and grown from the inside and outside surfaces of the RV, respectively.

The ASME Code, Section XI, Appendix G, methodology requires that applicants determine the limiting materials' adjusted reference temperature (ART). The ART is defined as the sum of 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 (M) term.

The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent on the amount of copper and nickel in the material and may be determined either from tables in RG 1.99 or from surveillance data. The fluence factor depends on the neutron fluence at the maximum postulated flaw depth. The margin term depends on whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF 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 calculational procedures. RG 1.99 describes the methodology for calculating the margin term.

Appendix H to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E185-82. Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine whether they meet the relevant requirements of Appendix H as they relate to determining and monitoring material fracture toughness.

In addition, RG 1.190 describes attributes of vessel fluence calculational methodologies (or equivalent) that are acceptable to the staff.

5.3.2.1 *Summary of Technical Information*

DCD Tier 2, Section 5.3, describes material properties and the effects of irradiation on material fracture toughness and the irradiation surveillance requirements. . DCD Tier 2, Section 5.3.3, outlines the vessel design bases for material construction, fabrication, inspection, operating conditions, inservice surveillance, safety design, and power generation. The section continues with a description of RV internals, CRD housing, in-core neutron flux monitoring, RV insulation, and RV nozzle design and inspections. The section concludes with fabrication methods, inspection requirements, and the 10 CFR 50.55a vessel requirements.

The ESBWR DCD discussion on P/T limits indicates that the heatup and cooldown P/T limit curves are required as a means of protecting the RV during startup and shutdown to minimize the possibility of brittle fracture. The P/T limits are based on flaw sizes specified in Subsection G-2120 of ASME Code, Section XI, Appendix G, "Protection Against Nonductile Failure." The maximum through-wall temperature gradient from continuous heating or cooling at 55.6 degrees C (100 degrees F) per hour was considered. The safety factors applied were those specified in ASME Code, Section XI, Appendix G. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the ART, which includes a reference nil ductility temperature shift, initial RT_{NDT}, and margin. The initial RT_{NDT} of the vessel materials is determined in accordance with the methodology presented in ASME Code, Section III, Subsection NB-2320; DCD Tier 2, Revision 9, Table 5.3.1, lists the requirements.

The applicant evaluated the RV flange, RV head and flange areas, FW nozzles, bottom head, and core beltline areas. The operating limit curves are based on the most limiting locations. The P/T limits are based on flaw sizes specified in Subsection G-2120 of ASME Code, Section XI, Appendix G. The analysis considered the maximum through-wall temperature gradient from continuous heating or cooling at 55.6 degrees C (100 degrees F) per hour. The safety factors applied were those specified in ASME Code, Section XI, Appendix G.

The applicant stated that the P/T curves are developed considering a radiation embrittlement of up to 54 EFPYs. This is consistent with an expected plant life of 60 years, with a 90-percent load factor. The maximum chemical compositions for the RV materials used in the core beltline region are as follows:

- Base materials: 0.05-percent maximum copper, 0.006-percent maximum phosphorus, 1-percent maximum nickel (for forging), and 0.73-percent nickel (for plate)

- Weld materials: 0.05-percent maximum copper, 0.008-percent maximum phosphorus, 1-percent maximum nickel, and 0.05-percent maximum vanadium

The projected RV fluence for the end of life is (E>1MeV):

- 1/4T location fluence: Less than 1.37×10^{19} neutrons per square centimeter (n/cm^2)

- 1/4T fluence for the weld above the top of the active fuel: 4.14×10^{17} n/cm^2

The operating curves are developed in accordance with Appendix G to 10 CFR Part 50. The initial RT_{NDT} for all RV materials is -20 degrees C (-4 degrees F). Thus, a minimum flange and boltup temperature of RT_{NDT} plus 33 degrees C (60 degrees F) or 13 degrees C (56 degrees F) will be used for tensioning at preload condition and during detensioning. In DCD Tier 2, Revision 9, Figures 5.3-1 and 5.3-2, the applicant provided generic curves for the ESBWR RV design. These are limiting curves based on the maximum copper and nickel contents and EOL peak fluence.

The results of the material surveillance program will verify the validity of ΔRT_{NDT} used in the calculation for the development of heatup and cooldown curves. The projected fluence, copper content, and nickel content, along with the RT_{NDT} calculation, will be occasionally adjusted, if necessary, using the surveillance capsules.

The applicant also indicated that temperature limits for core operation (both critical and noncritical), inservice leak tests, and hydrotests are calculated in accordance with ASME Code, Section XI, Appendix G.

5.3.2.2 *Staff Evaluation*

The staff reviewed the P/T limits for the ESBWR in accordance with SRP Section 5.3.2 to ensure that adequate safety margins existed for the structural integrity of the ferritic components of the RCPB.

In response to RAI 5.3-6, the applicant stated that the P/T calculation is performed in accordance with the requirements of Appendix G to 10 CFR Part 50. For the representative curves provided, the material initial RT_{NDT} data from the RV specification was used. To calculate the ART (accounting for the effects of irradiation in the vessel beltline region), the copper and nickel specification limits were used in combination with the peak fluence values and the methodology of RG 1.99. This is considered conservative since the actual RT_{NDT} values and chemical composition are normally much lower than the ones specified. Margins for the adjusted reference temperature calculation are consistent with those defined in RG 1.99.

DCD Section 5.3.2.1 states that for each individual component (e.g., main steam nozzle), a finite-element model was used to determine the stresses (pressure and thermal) for the transient events for normal and upset conditions. These stresses were then used to determine the applied K_I for each transient. The most limiting transient K_I for a given pressure and temperature was then compared to the minimum required K_{IC} (note that the minimum temperature limits of Appendix G to 10 CFR Part 50 also apply). The minimum required K_{IC} was based on the limiting RT_{NDT} of the materials for the component (determined as described above) and calculated using the methodology of ASME Code, Section XI , Appendix G. For the pressure test condition, a factor of 1.5 was applied to K_{IP} (K_I from primary membrane and bending stresses). For the core-not-critical and core-critical conditions, a factor of 2.0 was applied to K_{IP}. These safety factors are consistent with ASME Code, Section XI , Appendix G. Considering that the P/T limits described in the DCD are only representative and that plant-specific P/T limits will be provided during the COL application (see COL Information Item 16.0-1-A in DCD Section 5.3.1.5), the staff finds this approach acceptable.

The staff reviewed the P/T limits imposed based on the ESBWR RV materials to ensure that the P/T limits meet the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1), as they relate to the selection of materials for the RV and their ability to ensure adequate safety margins for the structural integrity of the RV ferritic components. SRP Section 5.3.2 indicates that P/T limits established for the RCPB must be consistent with the requirements of Appendix G to 10 CFR Part 50 and ASME Code, Section XI , Appendix G, to ensure satisfaction of the requirements for RCPB material fracture toughness. The applicant indicated that the temperature limits for core operation (both critical and noncritical), inservice leak tests, and hydrotests are calculated in accordance with Appendix G to 10 CFR Part 50 and ASME Code, Section XI , Appendix G. Thus, the probability of RV material failure and the subsequent effects on reactor core cooling and confinement is minimized. Therefore, the staff finds that the applicant has adequately met the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1).

The staff reviewed the P/T limits imposed on the RV to ensure that the materials selected for the RV meet the relevant requirements of GDC 14, in that they possess adequate fracture toughness properties to resist rapidly propagating failure and to act in a nonbrittle manner. The applicant indicated that the P/T limit curves will be developed in accordance with the criteria of Appendix G to 10 CFR Part 50, thereby ensuring a low probability of significant degradation or gross failure of the RV, which could cause a loss of reactor coolant inventory and a reduction in the capability to confine fission products.

The staff reviewed the RV materials to ensure that they meet the relevant requirements of GDC 31 as they relate to behavior in a nonbrittle manner and assure an extremely low probability of rapidly propagating fracture. In the DCD, the applicant indicated that RG 1.99 is used to calculate the ART. The staff finds this acceptable because RG 1.99 provides methods for predicting the effects of radiation on fracture toughness properties that are applicable to the requirements of GDC 31. In addition, the staff reviewed the P/T limits that will be imposed on the RCPB during preservice hydrostatic tests, inservice leak and hydrostatic tests, heatup and cooldown operations, and core-critical operation. The staff verified that adequate safety margins against nonductile behavior of rapidly propagating failure of ferritic components will exist, as required by GDC 31.

The staff reviewed the RV materials to ensure that they meet the relevant requirements of GDC 32 as they relate to the provision of a materials surveillance program. Compliance with Appendix H to 10 CFR Part 50 satisfies the requirements of GDC 32 for the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine whether they meet the relevant requirements of Appendix H to 10 CFR Part 50, as they relate to determining and monitoring fracture toughness. Section 5.3.2 of this report provides the staff’s review of the material surveillance program.

The applicant stated that the P/T limits are consistent with Appendix G to 10 CFR Part 50 and RG 1.99. The applicant also stated that it evaluated the vessel flange, RV head and flange areas, FW nozzles, bottom head, and the core beltline areas. The operating limit curves are based on the most limiting locations. The P/T limits are based on flaw sizes specified in Subsection G-2120 of ASME Code, Section XI, Appendix G.

The applicant confirmed that it performed the fluence analysis using the NRC-accepted methodology documented in the Licensing Topical Reports, NEDC-32983P-A, Class III (Proprietary), and NEDO-32983-A, Class I (Nonproprietary).

As stated above, the applicant provided P/T curves for the ESBWR design which are shown in DCD Tier 2, Revision 9, Figures 5.3-2 and 5.3-3. The DCD indicated that these curves are generic curves for the ESBWR RV design. In addition, they are the limiting curves based on the maximum copper and nickel material composition and EOL neutron fluence values. DCD Tier 2, Section 5.3.1.5, states that the COL applicant, in accordance with the TS (Chapter 16, Section 5.6.4), will furnish either bounding pressure and temperature curves as part of the TS or as part of a Pressure and Temperature Limits Report submittal for NRC review (COL Information Item 16.0-1-A).

In response to RAI 5.3-8, the applicant stated that the actual RV material properties will be used to refine the P/T curves before plant startup. The data from the surveillance capsules are available after plant startup in accordance with the schedule defined in DCD Tier 2, Revision 9, Section 5.3.1.6.1. Appendix H to 10 CFR Part 50, which the COL applicant is required to follow, defines the process to be followed if it is necessary to change the P/T curves based on the results of the surveillance program.

As required by 10 CFR 50.34(f)(2)(iii), which references TMI Action Item II.K.3.45, the vessel integrity limits must not be exceeded during rapid depressurization and rapid cooldown. The applicant stated that the ESBWR ADS DPVs are sized such that the vessel depressurization and cooldown are slow enough that vessel integrity limits are not exceeded. The applicant performed a comprehensive thermal hydraulic analysis that considered the effect of blowdown and reflooding by the GDCS. Hypothetical ESBWR accidents are calculated to be much slower than those of currently operating BWRs. In addition, it is expected that ESBWR operating procedures will be established so that actual transients will not be more severe than those for which the adequacy of the RV design has been demonstrated.

5.3.2.3 *Conclusions*

The staff concludes that the P/T limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure conform to the fracture toughness criteria of Appendix G to 10 CFR Part 50. A material surveillance program developed in conformance with Appendix H to 10 CFR Part 50 will determine the change in fracture toughness properties of the RV beltline materials during operation. The use of operating limits, as determined by the criteria defined in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur. This constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a; Appendix A to 10 CFR Part 50; and GDC 1, 14, 31, and 32.

5.3.3 Reactor Vessel Integrity

The staff reviewed DCD Tier 2, Section 5.3.3, Revision 9, in accordance with SRP Section 5.3.3, Revision 2. The applicant's assessment of RV integrity is acceptable if it meets codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure that the assessment meets the relevant requirements of 10 CFR 50.55a; Appendices G and H to 10 CFR Part 50; and GDC 1, 4, 14, 30, 31, and 32. These requirements are discussed below:

- GDC 1, GDC 30, and 10 CFR 50.55a(a)(1) require that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31 requires that the RCPB be designed with sufficient margin to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and will minimize the probability of rapidly propagating fracture.

- GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV.

- Appendix G to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G.

- Appendix H to 10 CFR Part 50 presents the requirements for a material surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E185. Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine that they meet the relevant requirements of Appendix H as they relate to determining and monitoring fracture toughness.

5.3.3.1 *Summary of Technical Information*

The RV is a vertical, cylindrical pressure vessel of welded low-alloy steel forging sections. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with ASME Code, Section III, Class 1 requirements. The ESBWR RV dimensions are as follows:

- Nominal inner diameter: 7.112 meters (m) (23.33 feet [ft])

- Nominal wall thickness including clad: 182 mm (7.17 in.)

- Minimum cladding thickness: 3.2 mm (0.125 in.)

- Nominal height from the inside of the bottom head (elevation zero) to the inside of the top head: 27.56 m (90.4 ft)

- Bottom of the active fuel location from elevation zero: 4405 mm (14.45 ft)

- Top of the active fuel location from elevation zero: 7453 mm (24.45 ft)

The cylindrical shell and top and bottom heads of the RV are fabricated of low-alloy steel, the interior of which is clad with stainless steel weld overlays, except for the top head and most nozzles. The main steam and bottom-head drain nozzles are clad with stainless steel weld overlay. The bottom head is clad with nickel-chromium-iron alloy.

A variety of welding processes, such as electroslag, SAW, manual welding, and automated GTAW, are used for cladding, depending on the location and configuration of the item in the vessel. Cladding in the “as-clad” condition may be acceptable for service if deposits are made with automatic processes, such as SAW, GTAW, and electroslag welding. For other processes, particularly where manual welding is employed, some grinding or machining is required.

Workmanship samples are prepared for each welding process in the “as-clad” condition and for typically ground surfaces.

The welding material used for cladding in the shell area is ASME Code, SFA-5.9 or SFA-5.4, type 309L or 309MoL, for the first layer, and type 308L or 309L/MoL for subsequent layers. For the bottom-head cladding, the welding material is ASME Code, SFA-5.14, type ERNiCr-3. DCD Tier 2, Revision 9, Table 5.2-4, lists the materials used in the RV.

The RV is designed and fabricated in accordance with the quality standards set forth in GDC 1 and 30 and 10 CFR 50.55a, as well as the requirements of Section III of the ASME Code. The design and construction of the RV enables inspection in accordance with Section XI of the ASME Code. In addition, the design documents impose additional requirements to ensure the integrity and safety of the RV. Design of the RV and its support system meets seismic Category I equipment requirements.

All plates, forgings, and bolting are 100-percent ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods, as required by ASME Code, Section III, Subsection NB. Welds on the RV are examined in accordance with methods prescribed in, and meet the acceptance requirements specified by, ASME Code, Section III, Subsection NB. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards provided in ASME Code, Section XI.

ISI of the RV must be performed in accordance with the requirements of Section XI of the ASME Code. The RV will be examined once before startup to satisfy the preoperational requirements of Subsection IWB-2000 of the ASME Code, Section XI. Subsequent ISI will be scheduled and performed in accordance with the requirements of 10 CFR 50.55a(g), as described in Section 5.2.4 of this report.

The material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the RV beltline region resulting from exposure to neutron irradiation and thermal environment. Specimens of actual reactor beltline material will be exposed in the RV and periodically withdrawn for impact testing. Operating procedures will be modified in accordance with test results to ensure brittle fracture control.

The RV support is considered a sliding support block type as defined in ASME Code, Section III, Subsection NF-3124. The vessel supports are constructed of low-alloy or carbon steel. Sliding supports are provided at a number of positions around the periphery of the vessel. The vessel support is designed to withstand the loading conditions specified in the design documents and meet the stress criteria of ASME Code, Section III, Subsection NF.

All piping connected to the RV nozzles has been designed not to exceed the allowable loads on any nozzle. Four drain nozzles are provided in the bottom head. Nozzles connecting to stainless steel piping have safe ends or extensions made of stainless steel. These safe ends or extensions are to be welded to the nozzles after the pressure vessel is heat treated to avoid furnace sensitization of the stainless steel. All nozzles, except the drain nozzles and the water-level instrumentation nozzles, are low-alloy steel forgings comprising ASME Code, SA-508, Grade 3, Class 1, material. The safe end materials used are compatible with the material of the mating pipes. The design of the nozzles conforms with ASME Code, Section III, Subsection NB, and meets the applicable requirements of the vessel design documents.

5.3.3.2 *Staff Evaluation*

Although the staff reviewed most areas separately in accordance with other SRP sections, the importance of the vessel integrity warranted a special summary review of all factors relating to RV integrity. The staff reviewed the fracture toughness of the ferritic materials for the RV, the P/T limits for the operation of the RV, and the materials surveillance program for the RV beltline. SRP Section 5.3.3 provides the acceptance criteria and references that form the bases for this evaluation.

The staff reviewed the information in each area to ensure that inconsistencies did not exist that would reduce the certainty of vessel integrity. The following is a list of the areas reviewed and the sections of this report in which they are discussed:

- RCPB materials (Section 5.2.3)
- RCS pressure boundary ISI and testing (Section 5.2.4)
- RV materials (Section 5.3.1)
- P/T limits (Section 5.3.2)

The integrity of the RV is ensured for the following reasons:

- The RV will be designed and fabricated to the high standards of quality required by the ASME Code and the pertinent ASME Code cases.

- The RV will be fabricated from material of controlled and demonstrated quality.

- The RV will be subjected to extensive PSI and testing to ensure that it will not fail because of material or fabrication deficiencies.

- The RV will operate under conditions, procedures, and protective devices that ensure that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients.

- The RV will be subjected to periodic inspection to demonstrate that its high initial quality has not deteriorated significantly under service conditions.

- The RV will be subjected to surveillance to monitor for neutron irradiation damage so that the operating limitations may be adjusted.

- The fracture toughness of the RV materials will be sufficient to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, they will behave in a nonbrittle manner and will minimize the probability of rapidly propagating fracture.

The ESBWR RV support is considered to be of a sliding support block type, as defined in ASME Code, Section III, Subsection NF-3124. These supports are not in the region of high neutron fluence, where neutron radiation embrittlement of the supports would be a significant concern. On the basis of the information provided, the staff considers the RV supports for the ESBWR design to be adequately designed to withstand the effects of radiation. Thus, the New Generic Issue 15, “Radiation Effects on Reactor Vessel Supports,” is resolved for the ESBWR design.

5.3.3.3 *Conclusions*

The staff finds that the structural integrity of the ESBWR RV meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices G and H to 10 CFR Part 50; and 10 CFR 50.55a. Therefore, the staff finds the structural integrity of the ESBWR RV to be acceptable. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the ESBWR plants conform to the applicable NRC regulations and RGs discussed above, as well as to the rules of Section III of the ASME Code. The ESBWR meets the fracture toughness requirements of the regulations and Section III of the ASME Code, including requirements for surveillance of vessel material properties throughout its service life, in accordance with Appendix H to 10 CFR Part 50. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G to ASME Code, Section III, and Appendix G to 10 CFR Part 50.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps Not Applicable to the ESBWR

5.4.2 Steam Generators Not Applicable to the ESBWR

5.4.3 Reactor Coolant Piping Not Applicable to the ESBWR

5.4.4 [Reserved]

5.4.5 [Reserved]

5.4.6 Isolation Condenser System

5.4.6.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Section 5.4.6, Revision 9, using relevant portions of SRP Section 5.4.6, draft Revision 4. Since the ICS is part of the ECCS, the staff also used SRP Section 6.3, Revision 3.

The staff compared the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, GIs, BLs, GLs, or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that draft Revision 4 of SRP Section 5.4.6 is acceptable for this review."

Acceptance criteria are based on the following:

- GDC 4, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer)

- GDC 5, "Sharing of structures, systems, and components," as it relates to SSCs important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair the ability of the SSCs to perform their safety function

- GDC 33, "Reactor coolant makeup," as it relates to the system's capability to provide reactor coolant makeup for protection against small breaks in the RCPB so that fuel design limits are not exceeded

- GDC 34, “Residual heat removal,” as it relates to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or RCPB overpressurization

- GDC 54, “Systems penetrating containment,” as it relates to the provision of leak detection and isolation capabilities for piping systems penetrating primary containment

- 10 CFR 50.63, as it relates to design provisions to support the plant’s ability to withstand and recover from a station blackout (SBO) of a specified duration

Section 6.3 of this report presents the acceptance criteria and the evaluation of the ICS as an ECCS.

5.4.6.2 *Summary of Technical Information*

DCD Tier 2, Sections 5.4.6.1 and 5.4.6.2, describe the ICS. The ICS removes decay heat after any reactor isolation during power operations. Decay heat removal limits additional pressure rise in the reactor and keeps the RPV pressure below the SRV pressure setpoint. The system consists of four independent loops, each containing a vertical heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the isolation condenser/passive containment cooling system (IC/PCCS) expansion pools, which are vented to the atmosphere.

To place an IC into operation, condensate return valves are opened, whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually from the MCR. Each IC has a fail-open nitrogen piston-operated condensate return bypass valve, which opens if the 250-volt direct current (dc) power is lost.

The IC/PCCS expansion pool is divided into subcompartments that are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC. The fuel and auxiliary pools cooling system (FAPCS) performs cooling and cleanup of IC/PCCS expansion pool water. During IC operation, IC/PCCS pool water can boil, and the steam produced is vented to the atmosphere.

ICs are capable of achieving and maintaining safe, stable conditions for at least 72 hours without operator action following non-LOCA events. Operator action is credited after 72 hours to refill IC/PCCS pools or initiate SDC.

The IC/PCCS pool has an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC/PCCS pool inventory. A safety-related FAPCS makeup line is provided to convey emergency makeup water into the IC/PCCS expansion pool from a water supply outside of the reactor building. The flowpath for this makeup can be established by manually opening the isolation valve on the FAPCS makeup line located at grade level in the yard area external to the reactor building.

The ICS passively removes sensible and core decay heat from the reactor (i.e., natural convection transfers heat from the IC tubes to the surrounding IC/PCCS expansion pool water, and no forced circulation equipment is required) when the normal heat removal system is unavailable.

The ICs are sized to remove postreactor isolation decay heat with three of four ICs operating and to reduce reactor pressure and temperature to safe-shutdown conditions (i.e., 216 degrees C [420 degrees F]), with occasional venting of radiolytically generated noncondensable gases to the suppression pool. The ICS operation is independent of station alternating current (ac) power and function whenever normal heat removal systems are unavailable to maintain reactor pressure and temperature below required limits.

The control room operators can perform periodic surveillance testing of the ICS valves by means of remote manual switches that actuate the isolation valves and the condensate return valves. Status lights on the valves verify the opening and closure of the valves. The essential monitored parameters for the IC/PCCS expansion pools are pool water level and pool radiation. IC/PCCS expansion pool water level monitoring is a function of the FAPCS. IC/PCCS expansion pool radiation monitoring is a function of the process radiation monitoring system.

5.4.6.3 *Staff Evaluation*

The staff assessed the design and function of the ESBWR ICS as described in DCD Tier 2, Section 5.4.6.

The ICS in the ESBWR is part of the ECCS and also serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main FWS is isolated from the RV. In addition, the ICS will provide decay heat removal necessary for coping with an SBO. The water supply for the ICS pools comes from the condensate demineralizer outlet or from the condensate storage tank.

The ICS is designed and qualified as a safety system. The ICS removes residual and decay heat from the reactor. The system employs natural circulation as the driving head through the IC vertical tubes. The shell side of the condenser is the IC/PCCS expansion pool, which boils off to remove heat transferred from the RCS. The heated IC/PCCS expansion pool vents to the outside atmosphere. This is achieved with minimal loss of coolant inventory from the reactor when the normal heat removal system is unavailable subsequent to any of the following events:

- Reactor isolation
- SBO
- ATWS
- LOCA

The IC operation automatically limits the reactor pressure and reduces the probability of SRV and ADS operation.

The staff reviewed the process diagram to verify that the essential ICS components are designated seismic Category I. The portions of the ICS (including isolation valves) that are located inside the containment and on the steamlines out to the IC flow restrictors are designed to ASME Code, Section III, Class 1, QG A, specifications. Other portions of the ICS are designed to ASME Code, Section III, Class 2, QG B, specifications. The IC/PCCS expansion pools are safety related and seismic Category I. Section 3.6.2 of this report discusses protection of the ECCS against pipe whip and discharging fluids (GDC 4). Section 3.11 of this report discusses environmental qualification of the ECCS equipment.

The ICS consists of four independent loops, and the ICS heat exchangers are sized to remove postreactor isolation decay heat with three out of four ICs operating (101.25 megawatt thermal

[MWt]) and to reduce reactor pressure and temperature to safe-shutdown conditions. Since the ICS design is capable of removing fission product decay heat and other residual heat from the reactor core (101.25 MWt), the system meets the requirements of GDC 34.

The IC, connected by piping to the RPV, is placed at an elevation above the RPV. When the steam is condensed, the condensate is returned to the vessel via a condensate return pipe. The steam-side connection between the RPV and the IC is usually open during normal operation. The accumulated subcooled water in the condensate return line is used for reactor coolant makeup during a LOCA.

Any of the following sets of signals generates an actuation signal for the ICS to come into operation:

- Two or more MSIV valve positions at less than or equal to 92-percent open, in separate main steam lines, with reactor mode switch in "run" only (percent-open values are those used in the safety analyses)

- RPV dome gauge pressure greater than or equal to 7.447 MPa (1,080 psig) for 10 seconds

- Reactor water level below Level 2, with time delay

- Reactor water below Level 1

- Loss of FW (loss of power to two-out-of-four FW pumps) with the reactor in the run mode

- Operator manual initiation

The condensate return line is provided with two parallel valves—an electro-hydraulic-operated, main valve, which fails as is, and a nitrogen piston-operated fail-open valve. This diversity provides more reliability for the system. Two normally closed, fail-closed, solenoid-operated lower head vent valves are located in the vent line from the lower headers. They can be actuated both automatically (when RPV pressure is high and either of the condensate return valves is open) or manually by the control room operator. A bypass line around the lower head vent valves contains one relief valve and one normally closed, fail-open solenoid valve. The valves are designed to open automatically at a pressure setpoint higher than that of the primary lower head vent valves. The vent line from the upper headers is provided with two normally closed, fail-closed, solenoid-operated upper header vent valves to permit opening of the noncondensable gas flowpath by the operator. All of the vent valves will be located in a vertical pipe run near the top of the containment. The vent lines will be sloped to the suppression pool to prevent accumulation of condensate in the piping. During ICS standby operation, discharge of potential entrained non-condensable gases or air is accomplished by a purge line that takes a small stream of gas from the top of the IC and vents it to the MSL. In RAI 5.4-32, S02, the staff asked the applicant for a detailed description of the nitrogen rotary motor-operated valve and the pneumatic piston-operated valve operation, including the actuator. The staff tracked RAI 5.4-32 as an open item in the SER with open items. In the response to RAI 5.4-32 S02, the applicant stated that the ESBWR design will have the option of using either gate valves or ball valves. The steamline isolation valves are nitrogen-powered piston valves and the condensate return valve actuators are electro-hydraulic operators, which use an electric motor (a pneumatic motor is used in place of the electric motor) driven pump to drive the piston. RAI 5.4-32 and its associated open item are resolved.

The four radiation monitors in the IC/PCCS expansion pool steam atmospheric exhaust passages for each IC loop are used to detect leakage from the IC outside the containment. Four sets of differential pressure transmitters are located in the steamline and the condensate return line to detect excessive flow as a result of a pipe break or a leak. The IC is isolated automatically when either a high radiation level in the IC pool area is detected or excess flow is detected in the steam supply line or condensate return line.

The IC/PCCS expansion pools have an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection can be continued indefinitely by replenishing the IC/PCCS expansion pool inventory. A safety-related, independent FAPCS makeup line is provided to convey emergency makeup water into the IC/PCCS expansion pool from the site fire protection system.

The ICS will have controls that will shut down the system if operating conditions exceed certain limits. The ICS is equipped with a leak detection system.

The vendor testing program, conducted at PANTHERS/IC test facility in Siet, Italy, included the IC performance test. The Siet test facility also performed full-scale testing on the simplified boiling-water reactor IC. The purpose of the prototype IC test was to show the ability of the IC heat exchanger to meet its design requirements for heat rejection (component performance). Section 21.5 of this report describes the staff evaluation of this vendor test.

Periodic heat removal capability testing of the ICS will be performed. During plant outages, routine ISI is required for the IC, including its piping, and supports, according to ASME Code requirements. The TS provide periodic test and surveillance requirements for essential components of the system. Chapter 14 of this report discusses the proposed initial test program. The ICS is also part of the ECCS, and Section 6.3 of this report describes the evaluation of the ECCS function of the system.

In RAI 5.4-37, the staff identified that the ICS and DPVs are connected to common stub lines from the RV. The staff raised concerns regarding the interaction between the ICS and the DPV when they were connected to the same stub tube in an earlier design stage. Early in an RCS depressurization, if the ICS is in operation, blowdown through the DPVs may draw fluid back from the cold side of the IC, as well as from the upper part of the RV. Depressurization loads will also affect the ICS, which serves as the primary boundary between the RCS and the environment, since the IC pools are outside of containment. Because the ICS is part of the ECCS, the ICS is expected to be physically separate from the DPVs, which are also part of the ECCS. The staff requested that the applicant (1) discuss the ramifications of the common tie between the ICS and the DPVs on the stub line from the RV, (2) explain why the design does not meet the physical separation criterion for the ECCS, and (3) describe in detail the potential system interactions and explain why there is no negative impact from the cross-tie between the IC steamline and the DPVs.

In response, the applicant supplied the following information:

> The cross-tie between IC steam line and DPVs in the ESBWR produces no significant negative impact on the loads and safety margins. The key details are as follow:
>
> 1. During a LOCA event, the peak operation of ICS occurs during the early part of the depressurization and before the DPV openings.

2. At the time of first DPV opening, there is no sub cooled water inside the IC drain line and in the down comer region. The total dynamic head (DPV flow + IC steam flow) inside the stub tube is small and will not induce backflow into the IC tubes.

3. Failure of one IC drain valve or one DPV valve will not prevent the operation of the other system connecting to the common stub line.

4. Based on (1) and (3), the common-tie between the ICS and DPVs on the stub line has no significant impact on the safety margins (refer to (5) below). Therefore, the physical separation of these two systems is not necessary.

5. Parametric studies were performed with and without the function of the IC heat transfer (i.e., no IC condensation). The results indicate that the long-term containment pressure is slightly higher for the case without the function of IC heat transfer.

The following paragraphs provide additional details on the system interactions as provided in RAI 5.4-37 response:

> The nozzles for the stub line and the IC drain line connect to the RPV at elevations of 21.9 m (71.9 ft) and 13.0 m (42.3 ft), respectively (reference to the RPV bottom). The bottom of IC tubes is approximately at 6 m (19.7 ft) above the stub line elevation, or approximately 15 m (42.2 ft) above the IC drain line nozzle elevation.
>
> In the early stages of RCS depressurization (0 - 500 seconds, before the opening of DPVs), the ICs are in operation and condense significant amount of steam flow from the RPV. For example for the MSL break case, they condense approximately 36 kg/s (79.4 pounds mass [lbm]/s) per IC. The steam flow to the ICS reduces as the RPV pressure decreases and the downcomer water level drops. The first group of ADS valves open after the downcomer level drops below the Level 1 set point (11.5 m [37.7 ft] from the RPV bottom, Table 6.3-1, DCD Rev. 2). Consequently, both the RPV pressure and the steam flow to the ICS reduce further after the first ADS valve opening. The first group of DPV valves opens at 50 seconds after the first ADS valve opening. At this time, the RPV pressure decreases to about 700 kPa (100 psia), the DPV flow is about 7.5 kg/s (16.5 lbm/s) per DPV and the IC steam flow reduces to about 4 kg/s (8.8 lbm/s) per IC. The total velocity inside the stub tube is in the range of 35 m/s (114.8 ft/s). The dynamic head is in the range of 2.2 kPa (0.3 psia), which is small compared to the static head of two-phase mixture in the vertical portion of the IC drain line.
>
> At the time of DPV opening, the RPV downcomer as well as the IC drain lines are filled with saturated two-phase mixture due to the fast depressurization resulting from the opening of ADS valves. As the result of additional depressurization from the DPV opening, the downcomer two-phase level could swell up a few meters from the Level 1.0 position, and get closer to or below the stub line elevation. However, there is no sub cooled water inside the IC drain line, or inside the downcomer near by the nozzle elevations of the IC drain line or the stub line. In addition, there are loop seals at the lowest elevation of the IC drain

lines, near by the injection nozzles. The loop seal provides extra static head, in addition to the 15 meters (49.2 ft) of static head of the two-phase mixture inside the vertical portion of the IC drain line, to prevent any flow reversal in the IC drain line and steam inlet line due to the DPV opening.

The applicant indicated that the information provided in response to RAI 5.4-37 S01 is included in DCD Tier 2, Revision 3, Sections 5.4.6.2.2 and 5.4.6.2.3. The staff determined that the information provided in DCD Tier 2, Revision 3, Sections 5.4.6.2.2 and 5.4.6.2.3 is adequate. Therefore, RAI 5.4-37 is considered resolved.

Section 6.2 of this report discusses containment isolation in accordance with the requirements of GDC 54. GDC 5 is not applicable because the ESBWR is a single-unit plant.

The condensate return line is sloped downward from the IC to an elevation below reactor water level to reduce the trapping and collapse of the steam in the drain piping. The staff believes that this sloping will reduce the potential for water hammer events during system startup.

The ICS is designed as a high-pressure reactor coolant makeup system that will start independent of the ac power supply. ICS heat exchangers are independent of plant ac power, and they function whenever normal heat removal systems are unavailable to maintain reactor pressure and temperature below limits. Subsequent to an SBO, the system is initiated when the RPV water Level 2 is reached following a trip of the FW pumps. The ICS initiates when the condensate return line valve opens using safety-related dc power.

The IC/PCCS expansion pool makeup serves as a clean water supply for replenishing the pool level during normal plant operation; the FAPCS provides level monitoring.

Because the materials selected for the IC are considered corrosion resistant, leakage across these components to the IC/PCCS pool is not expected. As a result, the prompt identification and response to leakage are important since the leakage indicates degradation of this barrier. In response to RAI 5.4-53, the applicant indicated that the alarm setpoint (in contrast to the high radiation setpoint) is selected close enough to background so that the alarm gives an early warning of a detected leak. In this response however, the applicant did not indicate the actions to be taken in response to such an IC radiation alarm (e.g., plant shutdown, inspection of the IC tubes), and did it not address why the leak rate associated with a critical size was not used in determining when the IC shall be isolated. The staff tracked RAI 5.4-53 as an open item in the SER with open items.

In RAI 5.4-53 S01, the staff requested that the applicant address the staff concerns in the open item discussed above. The applicant’s response stated that effluent radiation monitoring logic initiates an automatic isolation of the effected ICS division, requiring no immediate operator action. Followup actions to a radiation detector alarm are directed by the response procedure for the alarm, and operators will carry out actions in accordance with TS-based procedures. Operating plant procedures will address issues such as confirmation of the IC train isolation, investigation and determination of the cause for the isolation, development of a response plan, examination and repair of an IC heat exchanger (if required), or required actions for other equipment, and restoration of the train to operable status in accordance with the TS. With regard to the use of a critical flaw size in determining when the IC shall be isolated, the applicant stated that a critical flaw size is not a key parameter for determining the radiation monitor setpoint in order to maintain the health of any IC train or limit a release from the plant. The automatic IC train isolation at the radiation monitoring system alarm setpoint provides a limit for the rate of release to ensure that the site boundary radiation dose limits are not exceeded. Since the applicant described the actions to be taken in response to a radiation detector alarm as a result of an isolation condenser leak, and since the alarm setpoint is based on not exceeding site boundary radiation dose limits, the staff finds that the applicant has adequately addressed the concerns identified in RAI 5.4-53 S01. Therefore, RAI 5.4-53 S01 and associated open item are resolved. Additional information concerning radiation monitor alarm setpoints is located in Section 11.5 of this report.

In RAI 5.4-20, the staff requested that the applicant provide detailed information pertaining to the ICS design. In response to RAI 5.4-20, the applicant indicated that the IC tubes would be fabricated from a modified form of Alloy 600 (ASME Code Case N-580-1). However, in other portions of its submittal, the applicant stated that Alloy 600 would be used in the fabrication of the IC tubes. In this response, the applicant also indicated that the IC tubes would be bent by induction bending. However, the applicant did not indicate what effect, if any, this would have on the material properties of the tubing, and it did not indicate what testing, if any, would be performed to confirm the acceptability of the material properties following bending of the piping and tubing. In RAI 5.4-20 S01, the staff asked the applicant to clarify the actual type of Alloy 600 to be used in the IC. With regard to the applicant's discussion of induction bending of the IC tubes, the staff requested that the applicant discuss how it has confirmed that the material properties of the most limiting bent tube will remain acceptable following induction bending. The staff also asked the applicant to include a discussion of the material properties tested (e.g., hardness), the results, and the acceptance criteria.

In response to RAI 5.4-20 S01, the applicant indicated that the design of the support structures of the IC tubes was not currently available. The staff noted that, depending on the design, possible crevices between the IC tube and the support could result in the accumulation of chemical contaminants that could lead to corrosion. In addition, the materials of construction of the support are important in that they could corrode and result in a loss of support for, or damage to, the IC tubes. Because material selection and specific design attributes, such as the presence of crevices, can contribute to degradation, the staff requested, in RAI 5.4-20 S02, that the applicant provide a COL item to submit this information. The staff tracked RAI 5.4-20, related to the IC as an open item in the SER with open items. Section 6.1.1.3 of this report discusses the resolution of RAI 5.4-20. Based on the staff's evaluation, a COL Item was not required.

Table 6.1-1 of the DCD indicated that Alloy 600 would be used for IC tubing and header fabrication. Alloy 600 has a history of being susceptible to SCC in LWR systems. In RAI 6.1-10, the staff asked the applicant to provide a basis for the use of Alloy 600 in the IC, including material condition (i.e., mill annealed or thermally treated) as it relates to susceptibility to SCC in the reactor coolant and demineralized water environment.

The applicant indicated that there have been no reports of Alloy 600 cracking in BWRs in the absence of a welded crevice or a crack initiated in adjacent Alloy 182. These initiating features are absent from the ESBWR design. In addition, the material used for the IC is the same alloy as that used for the reactor shroud support and stub tubes (see applicant's response to RAI 4.5-18). This alloy (see ASME Code Case N-580-1) is a significantly modified version of Alloy 600, wherein the carbon content is limited, niobium (columbium) is added as a stabilizer, and high-temperature solution heat treatment is required instead of a mill anneal. Stress corrosion resistance is very good. The alloy is approved for use by the ASME Code (Code Case N-580-1) and has been deployed in several operating BWRs, including the Kashiwazaki-Kariwa 6/7 ABWRs. Several of these units have been operating for more than 10 years. In RAI 5.4-55, the staff requested that the applicant discuss the corrosion allowances for Alloy 600 used in the IC. The applicant responded that the Alloy 600 tubing in early BWR ICs performed satisfactorily without incident related to general corrosion in this application. Although general corrosion is not a concern, the applicant did not address whether any other incidences of corrosion or other degradation have occurred in operating units. The staff tracked RAI 5.4-55 as an open item in the SER with open items. Section 6.1.1.3 of this report discusses the resolution of RAIs 5.4-55 and 6.1-10.

In RAI 5.4-58, the staff requested that the applicant discuss any inspections and results of inspections of Alloy 600 in operating BWRs. In response to RAI 5.4-58, the applicant indicated that modified Alloy 600 has been in service for a number of years, but it is not currently inspected as part of a formal ISI program. In RAI 5.2-56, the staff asked the applicant to confirm that the method or technique for the inspection of IC tubes is capable of detecting general wall thinning, pit-like defects, and SCC along the entire length of the tube. In response to RAI 5.4-56, the applicant indicated that, because of the size of the IC tubes (2 nominal pipe size [NPS]), the IC tubes are exempted from volumetric and surface inservice examinations by ASME Code, Section XI, Subsection IWC-1220, which exempts sizes NPS 4 and smaller. The applicant contends that the ICs are subject to leakage (VT-2) examination under ASME Code, Section XI. Given the lack of long-term service experience (with inspection results) and the limitations of accelerated corrosion testing to fully simulate the range of variables that may exist in the field (and are pertinent to corrosion), the staff requested in RAI 5.4-58 S01 that the applicant provide additional information concerning the inspection and acceptance criteria for the IC tubes or justify why inspection requirements are not needed. The staff noted that the applicant's response to RAI 5.4-56 did not address the information requested by the staff. Therefore, RAI 5.4-58 S01 also requested that the applicant address the original issues posed in RAI 5.4-56. The staff tracked RAIs 5.4-56 and 5.4-58 as open items in the SER with open items. Section 6.6.3.3 of this report discusses the resolution of RAIs 5.4-56 and 5.4-58.

In response to RAI 5.4-47, the applicant stated that corrective maintenance for IC tube plugging following tube leak detection can be performed during refueling. After closing the isolation valves to and from the IC and after emptying its pool, personnel operating from the refueling floor can perform subcompartment plugging and repair of the leaking tube. Maintenance will be performed from the upper and lower end, after removal of the header covers. A remotely operated tool will be used to reduce radiation exposure to personnel. If there is considerable damage to some component part of the IC, each module of the IC unit is designed to be easily removable, after cutting the feed, drain, and vent lines. Also, the pool water in a specific IC subcompartment is designed to be removable without requiring the emptying of the remaining IC/PCCS expansion pools. The applicant also described the design features incorporated to reduce radiation exposure to personnel during IC tube plugging. The RAI 5.4-47 response is acceptable.

In response to RAI 5.4-51, the applicant stated that the ICS is designed to remove postreactor isolation decay heat with three out of four IC heat exchangers operating and to reduce the RCS temperature to safe-shutdown conditions of 204 degrees C (400 degrees F) in 36 hours with occasional venting to the suppression pool of radiolytically generated noncondensable gases. The ICS is capable of achieving and maintaining the safe-shutdown conditions without operator action for at least 72 hours. The safety-related flowpaths of the FAPCS are designed to provide makeup water beyond 72 hours to the ICS. The FAPCS has the ability to supply water to the ICS pools when connected to the fire protection system (FPS). Permanently installed piping is included in the FAPCS, which is connected directly with the site FPS, and this can provide makeup water from 72 hours through 7 days. The applicant adequately described in detail the use of ICS as requested in the RAI in combination with other systems to keep the plant in Safe Shutdown Condition for 72 hours in both normal shutdown mode as well as postaccident conditions. The applicant also described the use of ICS in combination with other systems during a LOCA and in post LOCA conditions. The RAI 5.4-51 response is acceptable.

Periodic heat removal capability testing of the IC is performed during normal plant operation at 5-year intervals.

On August 10, 2006, the applicant informed the staff that it was incorporating a change to the ICS drainline into the ESBWR design. According to the applicant, the reason for this change was to improve operator flexibility and to maintain minimum chimney collapsed level during a LOCA. This change also reduced the probability of ADS trip in SBO and loss-of-FW events. In addition, this change resulted in the elimination of the Level 1.5 trip and simplified ADS logic so that only a Level 1 setpoint was required. The new inline vessel (tank) is located on each ICS train condensate return line to provide the additional condensate volume for the RPV. The staff requested that the following information shall be added to the ICS ITAAC:

1. The calculated flow resistance in TRACG between the ICS condensate return line and the reactor

2. In ITAAC No. 20, the total volume assumed in the analysis for the IC/PCCS expansion pool

The staff requested this change in RAI 14.3-146. This RAI was tracked as an open item in the SER with open items.

In response to RAI 14.3-146, the applicant stated that ICS performance is not determined by controlling the drain line resistance. The heat removal capacity is the key safety significant parameter. The applicant revised the ITAAC 2.4.1 "Isolation Condenser System", including the heat removal capacity of the ICS and the total volume of the IC/PCCS expansion. Therefore, RAI 14.3-146 and its associated open item are resolved.

In addition, the staff requested, in RAI 5.4-22, S02, that the applicant discuss the means it will use to make certain that the ICS drainline is full during normal operation, thus ensuring that the water volume assumed in the safety analysis is available for injection upon a LOCA signal. The staff tracked RAI 5.4-22 as an open item in the SER with open items. In response to the RAI 5.4-22, S02, the applicant stated that a temperature element will be provided in each condensate return line downstream of the isolation valve and at the bottom and top of the condensate line at the RPV connection. Each temperature will be recorded in the MCR. The temperature measurements can provide information that the condensate line is filled with condensate. The staff was satisfied with the applicant's response. Therefore, RAI 5.4-22, S02 and associated open item are resolved.

DCD Tier 2, Revision 5, Section 5.4.5 deleted reference to GDC 54. The ICS steam supply and condensate lines penetrate the containment. GDC 54 is applicable to the ICS, as indicated in SRP Section 5.4.6, and therefore, should be included in the DCD. In response to RAI 5.4-63, S01, the applicant revised DCD Tier 2, Section 5.4.6 to include GDC 54; therefore, this issue is resolved.

Section 15.5.6 of this report discusses the ESBWR design's compliance with 10 CFR 50.63.

5.4.6.4 *Conclusions*

NRC has reviewed the applicant's information related to the IC system. The staff finds that the applicant has adequately demonstrated that the IC system is capable of decay heat removal during reactor isolation, SBO, and LOCA. The staff finds that the RWCU/SDC system meets the requirements of GDC 4, 5, 33, 34, 54 and 10 CFR 50.63.

5.4.7 Residual Heat Removal

5.4.7.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 5.4.7, in accordance with the staff position outlined in the applicable sections of SRP Section 5.4.7, draft Revision 4.

The staff compared the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, GIs, BLs, GLs, or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that draft Revision 4 of the SRP Section 5.4.7, is acceptable for this review."

Because of the functional limitations of the passive plant designs, the Commission, in a staff requirements memorandum (SRM) dated June 30, 1994, approved the position in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety systems in Passive Plant Designs," dated March 28, 1994. This position accepts a value of 215.6 degrees C (420 degrees F) or lower (rather than the cold shutdown specified in RG 1.139, "Guidance for Residual Heat Removal," issued for comment in May 1978) as the safe, stable condition that the passive systems must be capable of achieving and maintaining following non-LOCA events.

Acceptance criteria are based on the following:

- GDC 1, as it relates to the quality standards of the SSCs important to safety

- GDC 2, with respect to the seismic design of the system

- GDC 4, as it relates to the dynamic effects associated with flow instabilities and loads (e.g., water hammer)

- GDC 5, as it relates to SSCs important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair the ability of the SSCs to perform their safety function

- GDC 19, "Control room," as it relates to the provision of a control room from which actions can be taken to operate the nuclear power plant

- GDC 34, as it relates to the system design's capability to remove fission product decay heat and other residual heat from the reactor core to preclude fuel damage or RCPB overpressurization

5.4.7.2 *Summary of Technical Information*

The SDC mode of the RWCU system is the normal residual heat removal system for the ESBWR. The RWCU/SDC performs the following functions:

- Removal of decay heat during normal plant shutdowns
- Removal of the core decay heat, assuming either the main condenser or ICS is available for initial cool down
- With loss of preferred offsite ac power, bringing the plant to cold shutdown within 36 hours, in conjunction with the ICS, assuming the most restrictive single-active failure

In conjunction with the heat removal capacity of either the main condenser or the ICS or both, the RWCU/SDC system can reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below boiling at atmospheric pressure in less than 1 day. The system is also designed to control the reactor temperature reduction rate. The system can be connected to nonsafety-related standby ac power (standby diesel generators), which allows the system to fulfill its reactor cooling functions during conditions when the preferred power is unavailable.

The SDC function of the RWCU/SDC system provides decay heat removal capability at normal reactor operating pressure, as well as at lower reactor pressures. The redundant trains of the RWCU/SDC system permit SDC even if one train is out of service; however, cooldown time is extended when using only one train. If preferred power is lost, the RWCU/SDC system, in conjunction with the ICS, is capable of bringing the RPV to the cold shutdown condition in 36 hours, assuming the most limiting single-active failure, with the ICs removing the initial heat load.

The operation of the RWCU/SDC system at high reactor pressure reduces the plant's reliance on the main condenser or ICS. The entire cooldown is controlled automatically. As cooldown proceeds and reactor temperatures are reduced, pump speeds are increased and various bypass valves are opened, as described below.

During the early phase of shutdown, the RWCU/SDC pumps operate at reduced speed to control the cooldown rate to less than the maximum allowed RPV cooling rate. To maintain less than the maximum allowed RPV cooling rate, both RWCU/SDC trains are placed into operation early during the cooldown, with the pumps and system configuration aligned to provide a moderate system flow rate. The flow rate for each train is gradually increased as RPV temperature drops. To accomplish this, in each train, the bypass line around the regenerative heat exchanger (RHX) and the bypass line around the demineralizer are opened to obtain the quantity of system flow required for the ending condition of the SDC mode. In addition to the inlet valve to the nonregenerative heat exchanger (NRHX) of the reactor component cooling water system (RCCWS) being open, at an appropriate point, the motor-operated RCCWS inlet valve opens to increase the cooling water flow to each NRHX. The automatic reactor temperature control function governs the adjustable speed drive and controls the cooldown by gradually increasing the speed of the system pumps up to the maximum pump flow rate. Water purification operation continues without interruption. Over the final part of the cooldown, maximum flow is developed through the RWCU/SDC pumps. Flow rate reduction becomes possible while maintaining reactor coolant temperatures within target temperature ranges. CRD system flow is maintained to provide makeup water for the reactor coolant volume contraction that occurs as the reactor is cooled down. The RWCU/SDC system discharge line is used for fine-level control of the RPV water level as needed.

During hot standby, the RWCU/SDC system may be used, as required, in conjunction with the main condenser or IC to maintain a nearly constant reactor temperature by processing reactor coolant from the reactor bottom head and the midvessel region of the RV and transferring the decay heat to the RCCWS by operating both RWCU/SDC trains and returning the purified water to the reactor via the FW lines. The pumps and the instrumentation necessary to maintain hot standby conditions can be connected to the standby ac power supply during any loss of preferred power.

The RWCU/SDC system can be used to supplement the FAPCS spent fuel heat removal capacity during refueling (or at other times). The system also can provide additional cooling of the reactor well water when the RPV head is off in preparation for removing spent fuel from the core.

In conjunction with the ICs, the system has the capability of removing the core decay heat, plus drain excess makeup resulting from the CRD purge flow, 30 minutes following control rod insertion.

5.4.7.3 *Staff Evaluation*

The staff assessed the design and function of the RWCU/SDC for the ESBWR as described in DCD Tier 2, Revision 9, Section 5.4.7.

The ESBWR RWCU/SDC is a nonsafety-related system and is not required to operate to mitigate design-basis events. However, some of the valves of the RWCU/SDC perform the following safety-related isolation functions:

- Containment isolation of RWCU/SDC lines penetrating containment using containment isolation valves, according to the criteria specified in DCD Tier 2, Revision 9, Section 6.2.4.

- Preservation of the RCS pressure boundary integrity using pressure isolation valves, according to the criteria specified in DCD Tier 2, Revision 9, Section 5.4.8.

The RWCU/SDC is designed to remove both residual and sensible heat from the core and the RCS during shutdown operations, with the capability to (1) reduce the temperature of the RCS from 270 degrees C (518 degrees F) to 49 degrees C (120 degrees F) within 96 hours after shutdown in conjunction with the heat removal capacity of the main condenser or the ICs or both, and (2) maintain the reactor coolant temperature at 49 degrees C (120 degrees F) for the entire plant shutdown.

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, as well as in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, the staff specified that the ALWR designs shall reduce the possibility of a LOCA outside containment by designing, to the extent practical, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full reactor pressure. DCD Tier 2, Revision 9, Section 5.4.8.1.2, discusses the ESBWR design features that address the intersystem LOCA (ISLOCA). Section 5.4.8.1.2, states that "the supply side of the RWCU/SDC system is designed for the

RCPB design pressure plus 10 percent. Downstream of the pumps, the pump shutoff head at 5 percent overspeed is added to the supply side design pressure." The system is designed for operation at reactor pressure; therefore, the ISLOCA issue is resolved for the system.

In SECY-93-087, the staff specified that passive plants must have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance. The staff's review of the ESBWR design with respect to shutdown operations is based on the applicant's systematic assessment of shutdown operation concerns identified in NUREG–1449, "Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States," issued September 1993. DCD Tier 2, Revision 9, Section 19.4.7, provides this assessment. Section 19.2 of this report discusses the staff's evaluation of the shutdown operation issues. The present section addresses the issues raised in NUREG–1449.

Both RWCU/SDC adjustable speed drive pumps are connectable to the diesel generator bus during any loss of preferred power supply. There are two redundant trains, and the SDC has the capability to bring the reactor to cold shutdown conditions.

DCD Tier 2, Revision 9, Section 5.4.8.1.4, describes inspection and testing requirements for the SDC. Preoperational tests, which include valve inspection and testing, flow testing, and verification of heat removal capability, verify the proper operation of the SDC. The inspection and test requirements of the SDC valves are consistent with those identified in DCD Tier 2, Revision 9, Sections 5.2.4 and 6.2.6, respectively, for the valves that constitute the RCPB and the valves that isolate the line penetrating containment. In addition, DCD Tier 2, Revision 9, Table 6.2-31, includes these valves, which are subject to IST. The staff finds that the applicant has set proper inspection and test requirements for the SDC valves performing the safety-related functions of containment isolation and RCPB integrity preservation.

The design classifications of the RWCU/SDC components discussed above comply with GDC 1, which specifies that SSCs important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The RCPB portion and the containment isolation valves of the RWCU/SDC are designed as safety Class A. The remaining portions are designed as safety Class B. The system design meets Position C.2 of RG 1.29. This complies with GDC 2, which specifies that the SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes. Section 3.6.2 of this report discusses the protection of the RWCU/SDC system against pipe whip and against discharging fluids (GDC 4). GDC 5 is not applicable to the ESBWR design because the RWCU/SDC system is designed for a single nuclear power unit and is not designed to be shared between units. The RWCU/SDC system is operated from the MCR, thus satisfying the requirements of GDC 19. Because the RWCU/SDC system is not designed to provide safety-related heat removal mitigation of design-basis events, the safety-related ICS complies with the heat removal function of GDC 34.

Safe Shutdown

Establishing a safe-shutdown condition requires maintaining the reactor in a subcritical condition and providing adequate cooling to remove residual heat. One of the functional requirements for the ESBWR is that the plant can be brought to a stable condition using the safety-grade systems for all events. The Commission, in an SRM dated June 30, 1994, approved the position proposed in SECY-94-084. This position accepts temperatures of 215.6 degrees C

(420 degrees F) or below, rather than the cold shutdown temperature (less than 93.3 degrees C [200 degrees F]) specified in SRP Section 5.4.7, Branch Technical Position RSB 5-1, Rev. 4, 1996 as the safe, stable condition that the passive decay heat removal system must be capable of achieving and maintaining following non-LOCA events. The SLCS establishes safe shutdown by providing the necessary reactivity control to maintain the core in a subcritical condition and by providing residual heat removal capability to maintain adequate core cooling. DCD Tier 2, Revision 9, Section 7.4, discusses the systems required for safe shutdown.

For all events, the following systems are used to keep the reactor in a stable condition:

- ICs
- SLCS
- SRVs
- DPVs
- GDCS
- PCCS

The staff finds that the applicant is following the Commission's guidance in SECY-94-084 regarding safe shutdown aspects of the passive plant; therefore, the use of this approach is acceptable.

5.4.7.4 *Conclusions*

NRC has reviewed the applicant's information related to the RWCU/SDC system. The staff finds that the applicant has adequately demonstrated that the RWCU/SDC system has the capability to cool the RCS following shutdown and provide decay heat removal. The staff further finds that the RWCU/SDC system meets the requirements of GDC 1, 2, 4, 5, and 19.

5.4.8 Reactor Water Cleanup/Shutdown Cooling System

5.4.8.1 *Regulatory Criteria*

The staff reviewed the RWCU/SDC system description in accordance with SRP Section 5.4.8, Revision 3. Staff acceptance of the design is based on compliance with the requirements of GDC 1, 2, 14, 60, "Control of releases of radioactive materials to the environment," and 61, "Fuel storage and handling and radioactivity control." These requirements are discussed below:

- GDC 1, as it relates to the design of the reactor water cleanup system (RWCU) and components to standards commensurate with the importance of the system's safety function

- GDC 2, as it relates to the RWCU being able to withstand the effects of natural phenomena

- GDC 14, as it relates to ensuring the RCPB integrity

- GDC 60, as it relates to the capability of the RWCU to control the release of radioactive effluents to the environment

- GDC 61, as it relates to designing the RWCU with appropriate confinement

RG 1.56, Revision 1, "Maintenance of Water Purity in Boiling Water Reactors," issued July 1978, describes a method acceptable to the staff for implementing the criteria for minimizing the probability of corrosion-induced failure of the RCPB in BWRs by maintaining acceptable purity levels in the reactor coolant. It also describes instrumentation acceptable to the staff for determining the condition of reactor coolant and the coolant purification system.

5.4.8.2 *Summary of Technical Information*

The two basic functions of the RWCU/SDC system are reactor water cleanup and shutdown cooling. The RWCU/SDC system performs the reactor water cleanup function during startup, normal power operation, cooldown, and shutdown. The SDC function of the RWCU/SDC system provides decay heat removal capability in conjunction with the main condenser or the ICS at normal reactor operating pressure, as well as at lower reactor pressures. There are two redundant RWCU/SDC trains. Section 5.4.7 of this report presents the review of the SDC function of the RWCU/SDC system.

The RWCU/SDC system consists of the following major components:

- Demineralizers
- Valves and piping
- RHXs
- NRHXs
- Pumps with adjustable speed motor drives

The RWCU/SDC system functions are not safety related; therefore, the system has no safety-related design basis other then to provide a containment isolation function and instrumentation for detection of system breaks outside the containment.

5.4.8.3 *Staff Evaluation*

The staff reviewed the RWCU/SDC system description in accordance with SRP Section 5.4.8. Staff acceptance of the design is based on compliance with the requirements of (1) GDC 1, as it relates to the design's ability to meet standards commensurate with the system's safety function, (2) GDC 2, as it relates to the system being able to withstand the effects of natural phenomena, (3) GDC 14, as it relates to assuring the integrity of the RCPB, (4) GDC 60, as it relates to the capability of the system to control the release of radioactive effluents to the environment, and (5) GDC 61, as it relates to designing the system with appropriate confinement.

The RWCU/SDC system performs the following functions:

- Removes solid and dissolved impurities from the reactor coolant and measures the reactor water conductivity during all modes of operation, in accordance with RG 1.56 and Electric Power Research Institute (EPRI), "BWR VIP-130, BWR Vessel and Internals Project BWR Water Chemistry Guidelines" (BWR VIP-130)

- Discharges excess reactor water during startup, shutdown, and hot standby conditions and during refueling to the main condenser or to the radwaste system

- Minimizes RPV temperature gradients by enhancing circulation through the bottom head region of the RPV and reducing core thermal stratification at low power

- Provides containment isolation, which ensures that the major portion of the system is outside the RCPB

- Provides heated primary coolant for RPV hydrostatic tests and reactor startups

- Supplies redundant cleanup capacity for major system components

The RWCU/SDC system is a closed-loop system consisting of two independent trains. Each train consists of an RHX, an NRHX, a demineralizer, two circulating pumps, isolation valves, piping, and instrumentation. The system takes its suction from the midvessel area of the RPV and from the reactor bottom head and discharges back to the vessel via the FW lines. Incoming water is cooled by flowing through the tube side of the RHX and the NRHX before pump suction. After the NRXH, water moves through the demineralizer to remove all impurities, reheats the incoming reactor water via the shell side of the RHX, and returns to the RCS. Each train is capable of performing the functions of reactor water cleanup and SDC. The system capacity is 1 percent of the rated FW flow rate.

The system is classified as nonsafety-related with the exception of the containment isolation valves. The two independent trains are located in the reactor building. System piping from the RPV to the outboard containment isolation valve forms part of the RCPB and is classified as QG A; ASME Code, Section III, Class 1; and seismic Category I. In the remainder of the system downstream of the containment isolation valves, the piping is classified as QG C; ASME Code, Section III, Class 3; and seismic Category I. The RWCU/SDC return line from the isolation valve, up to and including the connection to the FW line, is classified as QG B; ASME Code, Section III, Class 2; and seismic Category I.

In RAI 5.4-7, the staff asked the applicant to provide the basis for designing the return line from the isolation valve, up to and including the connection to the FW line, as QG B. In response, the applicant stated that the portion of the RWCU/SDC system return line from the isolation valve to the interface with the FW line is designed to QG B to be consistent with the QG of the FW line at the interface. The staff finds the applicant’s response acceptable because it is consistent with RG 1.26. Therefore, the staff considers RAI 5.4-7 resolved.

In RAI 5.4-8, the staff asked the applicant to explain how the effects of high- and moderate-energy piping failures outside the primary containment were evaluated in the RWCU/SDC design to ensure that the other safety-related systems and equipment will not be made inoperable. In response, the applicant stated that DCD Tier 2, Sections 3.6.1.2 and 3.6.2.1, describe protection against dynamic effects associated with postulated rupture of piping outside the containment for high- and moderate-energy piping. The description includes the identification of the high-energy piping located outside the containment, the potential damage resulting from dynamic effects, the design-basis compartment break, compartment pressurization, and equipment qualification. Sections 3.6.1 and 3.6.2 of this report present the staff evaluation of these DCD sections. The staff finds the applicant’s response acceptable because the requirements of GDC 2 and GDC 4, as related to SSCs important to safety, are satisfied; therefore, the staff considers RAI 5.4-8 resolved.

In RAI 5.4-9, the staff asked the applicant to demonstrate the capability of safety-related systems to withstand the effects of postulated internally generated missiles from the RWCU/SDC system both inside and outside the primary containment. In response, the applicant stated that DCD Tier 2, Section 3.5.1, includes the evaluation of the ability of the safety-related systems to withstand the effects of internally generated missiles both inside and outside containment. Section 3.5.1 of this report discusses the staff's evaluation of this DCD section. Therefore, the staff finds the applicant's response acceptable because GDC 4 requirements in regard to SSCs important to safety are satisfied. Thus, RAI 5.4-9 is resolved. In addition, in RAI 5.4-10, the staff asked the applicant to demonstrate the capability of structures housing the RWCU/SDC, including safety-related components and instruments inside these structures, to withstand external and internal flood conditions. In response, the applicant stated that the RWCU/SDC system components are housed in the containment and the reactor building. DCD Tier 2, Section 3.4 describes the internal and external flooding evaluation. As a result of its response to RAI 5.4-10, the applicant revised DCD Tier 2, Sections 3.4.1.3, 3.4.1.4, and 3.4.1.4.2, to further clarify its flooding analysis. The staff finds the applicant's response acceptable because the requirements of GDC 2 and GDC 4 are satisfied and confirmed the changes in DCD Tier 2, Revision 2. Section 3.4.1 of this report presents the staff's evaluation of these DCD sections. Therefore, the staff considers RAI 5.4-10 resolved.

Based on this seismic and QG classification design information, the staff finds that, by following the guidelines of RGs 1.26 and 1.29, the applicant has met the requirements of GDC 1 and 2 as they relate to the ability of the RWCU/SDC design to meet standards commensurate with the system's safety function and to withstand the effects of natural phenomena.

The two safety-related containment isolation valves on the suction lines of the RWCU/SDC system receive isolation signals from the leak detection and isolation system. These valves will automatically isolate on the following indications:

- High RWCU/SDC flow
- Low reactor water level (Level 2)
- High temperature in the MSL tunnel
- Initiation of the SLCS

The suction lines of each train are isolated by one automatic nitrogen-operated gate valve inside and one air-operated gate valve outside the containment. The reactor bottom suction line has a sampling line isolated by one automatic nitrogen-operated globe valve inside and one air-operated globe valve outside the containment. RWCU/SDC pumps, heat exchangers, and demineralizers are located outside the containment. In addition, DCD Tier 2, Revision 9, Section 5.4.8.1.1, states that the RWCU/SDC meets the guidance of RG 1.56 and the EPRI report BWR VIP-130. Based on this information, supplemented by the seismic and QG classifications for the portion of the system including the containment isolation valves discussed above, the staff finds that the system meets the requirements of GDC 14 as it relates to assuring the integrity of the RCPB.

In RAI 5.4-4, the staff asked the applicant to describe the design features of the RWCU/SDC system that will control the release of radioactive effluents to the environment in accordance with GDC 60. In response, the applicant stated that contaminated liquid waste will be transferred to the liquid waste management system (LWMS). In addition, flushing connections are provided to decontaminate piping and equipment such as the demineralizers and the heat exchangers. The RWCU/SDC system is provided with piping connections routed to the main condenser and the LWMS. The piping has butt-welded connections, rather than socket welds, to reduce crud traps. If high radiation is detected downstream of the demineralizer, the flow will be manually shifted to the LWMS by first opening the remote manual isolation valve to the LWMS and then closing the remote manual system isolation valve to the main condenser. The staff finds the applicant's response acceptable because the purpose of the LWMS is to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including AOOs. Based on this information, the staff finds that the system meets the requirements of GDC 60 as it relates to the capability of the RWCU to control the release of radioactive effluents to the environment. The staff finds RAI 5.4-4 to be resolved.

The demineralizers used are of the mixed-bed type with nonregeneration-type resin beads. A resin strainer capable of removing resin particles from the demineralizers' effluent is located at the outlet line to prevent resin beads from entering the system. Resin bed performance is monitored by the process sampling system. Sample probes are located in the inlet and outlet lines of the two demineralizers where samples are routed to the sample station for analysis. In addition, the conductivity of the demineralizer influent and effluent streams is continuously measured and transmitted to the MCR. The performance of the resin beads determines their replacement time. Since nonregeneration-type resin beads are used, whenever it is necessary to replace the spent resins, the resin vessel will be isolated from the rest of the system before resin addition. In RAI 5.4-3, the staff asked the applicant to describe the resin transfer system and indicate the provisions taken to ensure that transfers are complete and that crud traps in transfer lines are eliminated. In response, the applicant stated that the details of the resin transfer system will be designed in the detail design phase and that it would add the following design description in DCD Tier 2, Section 5.4.8.1.2:

> The resin transfer system will be designed to prevent resin traps in sluice lines. Consideration will be given in the design to avoid collection of resins in valves, low points and stagnant areas.

The applicant committed to placing this statement in a future revision of the DCD. The staff finds the applicant's response acceptable and confirmed the changes in DCD Revision 5. The staff considers RAI 5.4-3 resolved.

Spent resins will be sluiced to a backwash-receiving tank from which they will be transferred to the radwaste system for processing and disposal. Demineralizers are located in separate concrete-shielded cubicles that are accessible through shielded hatches. Valves and piping within the cubicles are reduced to the extent that entry into the cubicles is not required during any operational phase. Most of the valves and piping are located in a shielded valve gallery adjacent to the demineralizer cubicles. The valves are remotely operable to the greatest practical extent to minimize entry requirements into this area. The backwash tank is shielded separately from the resin transfer pump.

Each demineralizer is protected from high flow, high differential pressure across the strainer and across the demineralizer, and from demineralizer inlet high temperature by a bypass valve. In the event of high differential pressure or high temperature, an alarm will be activated in the MCR to alert the plant operator. Alarm logic will automatically isolate the demineralizer by first opening the bypass valve and then closing the demineralizer inlet valve.

In RAI 5.4-5, the staff asked the applicant to describe the control features that will prevent inadvertent opening of the demineralizer backwash valves during normal operation. In response, the applicant stated that interlocks are provided to prevent inadvertent opening of the resin addition and back-flushing valves during normal operation. The staff finds the applicant's response acceptable because the use of interlocks will be adequate to prevent the inadvertent opening of the valves. The staff finds RAI 5.4-5 to be resolved.

SRP Section 5.4.8 states that, to prevent resin loss from the demineralizer bed, the RWCU system shall include a means for automatically maintaining flow through demineralizer beds in the event of low-process flow or loss of flow. In RAI 5.4-2, the staff asked the applicant to describe design requirements for a system controlling the ability of the demineralizer to automatically adjust flow through its resin beds to prevent resin loss in the event of a decrease of system flow. In response, the applicant stated that this SRP requirement does not apply to the ESBWR demineralizers because they use nonregeneration bead-type resins which do not lose resins on a reduction or loss of process flow. The staff finds the applicant's response acceptable and considers RAI 5.4-2 resolved.

The RHX and the NRHX are other components of the RWCU/SDC system that are exposed to high-radiation levels. These components are also located in shielded cubicles with valves operated remotely by use of extension valve stems or from instrument panels located outside the cubicle.

The cleanup flow leaving the NRHX and going into the demineralizers should be of a specific temperature; therefore, the NRHX should be able to maintain the required temperature of the cleanup flow when its cooling capacity is reduced as a result of partially bypassing a portion of the return flow to the main condenser or the radwaste system. In RAI 5.4-1, the staff asked the applicant to describe whether the NRHX has the capacity of maintaining the desired temperature when its return flow is reduced. In response, the applicant stated that the NRHX performance was evaluated in the cleanup mode with a reduced RHX capacity by assuming that 25 percent of its normal return flow is bypassed to the main condenser. Since the NRHX cools the reactor water by transferring heat to the RCCWS, increasing the water flow of the RCCWS will provide enough cooling capacity to maintain the required temperature of the cleanup flow to the demineralizer. This proved sufficient to maintain the demineralizer's required inlet temperature. The staff finds the applicant's response acceptable. Therefore, RAI 5.4-1 is resolved.

Based on this information, the staff finds that the RWCU/SDC system design meets the requirements of GDC 61 as it relates to designing the system with adequate confinement features in regard to minimizing the probability of releasing radioactive materials during normal operation and AOOs.

5.4.8.4 *Conclusions*

The RWCU/SDC system will be used to maintain the reactor water purity and to reduce the reactor water inventory as required by plant operations. The staff's review has included system schematics along with descriptive information concerning the system design and operation.

The staff finds that the proposed design of the RWCU/SDC system is acceptable and meets the relevant requirements of GDC 1, 2, 14, 60, and 61. This conclusion is based on the following:

- The applicant has met the requirements of GDC 1 by designing, in accordance with the guidelines of RG 1.26, the portion of the RWCU/SDC extending from the RV and recirculation loops to the outermost primary containment isolation valves to QG A and by designing, in accordance with Position C.2 of RG 1.26, the remainder of the system outside the primary containment to QG C.

- The applicant has met the requirements of GDC 2 by designing, in accordance with Positions C.1, C.2, C.3, and C.4 of RG 1.29, the portion of the RWCU/SDC extending from the RV and recirculation loops to the outermost primary containment isolation valves to seismic Category I.

- The applicant has met the requirements of GDC 14 by meeting the positions of RG 1.56 and BWR VIP-130 in maintaining reactor water purity and material compatibility to reduce corrosion probabilities, thus reducing the probability of RCPB failure.

- The applicant has met the requirements of GDC 60 and 61 by designing a system containing radioactivity with confinement and by venting and collecting drainage from the RWCU/SDC components through closed systems.

Based on this information, the staff finds that the RWCU/SDC design for the ESBWR is acceptable.

5.4.9 Main Steamlines and Feedwater Piping

The applicant provided information regarding MSL and FW piping in DCD Tier 2, Revision 9, Section 5.4.9. Chapter 10 of this report presents the staff's evaluation of these systems.

5.4.10 Pressurizer—Not Applicable to the ESBWR

5.4.11 Pressurizer Relief Discharge System—Not Applicable to the ESBWR

5.4.12 Reactor Coolant High-Point Vents

5.4.12.1 *Regulatory Criteria*

The staff performed its review of the ESBWR RCS high-point vent system in accordance with SRP Section 5.4.12, draft Revision 1, issued in 1996.

The staff compared the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, GIs, BLs, GLs, or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that SRP, draft Revision 1, Section 5.4.12, is acceptable for this review."

The following requirements appear in 10 CFR 50.34(f)(2)(VI):

> Provide the capability of high point venting of non-condensable gases from the RCS, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of LOCA or an unacceptable challenge to containment integrity (II.B.1).

Acceptance criteria are based on the following:

- 10 CFR 50.55a and GDC 1 and 30, as they relate to the vent system components that are part of the RCPB being designed, fabricated, erected, and tested and maintained to high quality standards

- GDC 14, as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- 10 CFR 50.46(b), as it relates to the long-term cooling of the core following any calculated successful initial operation of the ECCS to remove decay heat for an extended period of time

- 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” with respect to environmental qualification of electrical equipment necessary to operate the reactor coolant vent system

- GDC 17, “Electric power systems,” with respect to the provision of normal and emergency power for the vent system components

- GDC 19, with respect to the vent system controls being operable from the control room

- GDC 36, “Inspection of emergency core cooling system,” as it relates to the vent system being designed to permit periodic inspection

5.4.12.2 *Technical Information*

The ESBWR has an RPV head vent system that handles any noncondensable gas buildup at the high point inside the RPV head by sweeping the gases into a MSL and then ultimately to the condenser. Additionally, systems that are connected to the RPV and are stagnant during normal plant operation have lines that are sloped to prevent any buildup of noncondensable gases.

During reactor operation, the noncondensable gases that may collect in the reactor head and the ICS steamlines are drawn to the steamline through a vent line with two normally open motor-operated valves that goes from the RPV head to the MSL and a purge line that goes from each of the ICs to a MSL. Differential pressure between the reactor head and the downstream steamline location extracts the noncondensables. The noncondensables are swept from these lines to the condenser, where they are extracted. These vents and purge lines are not required to ensure natural circulation core cooling. The vent line used to vent the reactor head noncondensables following a refueling operation is isolated with two normally closed valves during reactor power operation. The ICs also vent noncondensables to the suppression pool to maintain ICS performance; however, the ICs are isolable and not part of the primary system. Section 5.4.6 of this report discusses the ICS vents.

5.4.12.3 *Staff Evaluation*

The staff reviewed the design and function of the RPV vent system, as described in DCD Tier 2, Revision 9, Section 5.4.12,.

The ESBWR meets the requirements of 10 CFR 50.34(f)(2)(VI), which references TMI Action Item II.B.1, regarding the capability of high-point venting of noncondensable gases from the RCS. The noncondensables are swept from the steamlines to the condenser, where they are extracted. Position indication and controls for opening and closing the valves are in the control room. These vents and purge lines are not required to ensure natural circulation core cooling. The staff reviewed the procedure for operation of the RPV head vent system information provided in DCD Tier 2, Section 5.4.12.1 and finds it to be acceptable.

When the RPV is in an isolated condition, the RPV head vent line and the SRVs provide redundancy for venting the RCS. The vent line used to vent the reactor head noncondensables following a refueling operation is isolated with two normally closed valves during reactor power operation. These valves are subject to an environmental qualification (10 CFR 50.49(a)) program, as described in DCD Tier 2, Revision 9, Section 3.11 and evaluated in Section 3.11 of this report.

GDC 17 is met by an onsite electric power system that provides normal and emergency power to permit operation of the RPV head vent line valves. GDC 19 is met by controls and indication that permit operation of the valves from the MCR. The RPV head vent system is not part of the ECCS and is not required to ensure natural circulation core cooling. Therefore, GDC 36 does not apply. For RCPB isolation purposes during reactor power operation, the use of two nitrogen-operated valves in series in the piping that vents the RPV to the equipment and floor drain sump provides redundancy. Either or both valves isolate the piping. Failure modes consist of loss of power supply, failure of the control system, and mechanical failure in the valve. If one of the valves experiences a failure, the second valve in series performs the isolation function. Indication of open and closed position, and of temperature downstream of the second valve are available to operators in the MCR.

A connection at the RPV flange area links the internal integral head vent piping to the external head vent piping. The piping is 2 inches in diameter. The vent piping directs air and noncondensable gases from the RPV to either the equipment and floor drain sump or one of the MSLs. The vent piping permits air to be released from the RPV so that the vessel can be filled with water for hydrostatic testing, vents gases during reactor operation and reactor shutdown, and provides the upper tap for RPV-level measurement during reactor shutdown. The diameter of the vent line piping is much smaller than the diameter of the MSL piping. Therefore, a break in this piping is bounded (in accordance with 10 CFR 50.46a) by an MSL break, which is addressed in DCD Tier 2, Revision 9, Section 6.3.

5.4.12.4 *Conclusions*

As discussed above, the RPV vent system design for the ESBWR complies with the guidelines of SRP Section 5.4.12 and therefore is acceptable. The staff finds that the design of the RCS high-point vents is acceptable because it meets the relevant requirements of 10 CFR 50.34(f)(2)(vi), 10 CFR 50.46a, 10 CFR 50.49, 10 CFR 50.55a, TMI-2 Action Item II.B.1, and GDC 1, 14, 17, and 19. The staff finds that the ESBWR design provides various means to prevent accumulation of noncondensable gases in the RCS.

6.0 ENGINEERED SAFETY FEATURES

6.1 <u>Engineered Safety Features Materials</u>

6.1.1 Engineered Safety Features Metallic Materials

6.1.1.1 *Regulatory Criteria*

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 6.1.1, in accordance with U.S. Nuclear Regulatory Commission (NRC), NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” March 2007 (hereafter referred to as the SRP). In the economic simplified boiling-water reactor (ESBWR) design control document (DCD), Tier 2, Revision 9, Section 6.1.1, the applicant described the selection, fabrication, and compatibility of materials with core cooling water and containment sprays for engineered safety feature (ESF) systems. The NRC staff (staff) based its review of DCD Tier 2, Revision 9, Section 6.1.1, and its acceptance criteria on the relevant requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a; Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities;” General Design Criteria (GDC) 1, 4, 14, 31, 35, and 41; and Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50.

- GDC 1, “Quality standards and records,” and 10 CFR 50.55a(a)(1) require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions they perform.

- GDC 4, “Environmental and dynamic effects design bases,” requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (e.g., loss-of-coolant accidents [LOCAs]).

- GDC 14, “Reactor coolant pressure boundary,” requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31, “Fracture prevention of reactor coolant pressure boundary,” requires that the design of the RCPB include sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and the probability of rapidly propagating fracture will be minimized.

- GDC 35, “Emergency core cooling,” requires a system to provide abundant emergency core cooling. GDC 35 also requires that, during activation of the system, clad metal-water reaction will be limited to negligible amounts.

- GDC 41, “Containment atmosphere cleanup,” requires that the design provide containment atmosphere cleanup systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment. The staff limited its review of the ESF structural materials to ensuring that they meet the requirements of GDC 41 with respect to corrosion rates related to hydrogen generation in postaccident conditions.

- Appendix B to 10 CFR Part 50 mandates that applicants establish quality assurance (QA) requirements for the design, construction, and prevention or mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

6.1.1.2 *Summary of Technical Information*

The ESFs of the ESBWR design are those systems provided to mitigate the consequences of postulated accidents. DCD Tier 2, Chapter 6, identifies the ESF systems, which include (1) fission product containment and containment cooling systems, (2) emergency core cooling systems (ECCSs), and (3) control room habitability systems.

The applicant has provided a Tier 2 description of the ESF systems materials in DCD Tier 2, Revision 9, Section 6.1.1, summarized here in part as follows:

The applicant stated that materials used in the ESF components have been evaluated to prevent material interactions that could potentially impair operation of the ESFs.

The applicant selected materials to withstand the environmental conditions encountered during normal operation and postulated accidents. The applicant considered the materials' compatibility with core and containment spray water and also evaluated the effects of radiolytic decomposition products.

The design uses primarily metallic and metal-encapsulated insulation inside the ESBWR containment. All nonmetallic thermal insulation must have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride, consistent with Regulatory Guide (RG) 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," dated February 23, 1973, to minimize the possible contribution to stress-corrosion cracking (SCC) of austenitic stainless steel.

DCD Tier 2, Revision 9, Section 5.2.3, provides the evaluation of RCPB materials, and DCD Tier 2, Revision 9, Table 5.2-4, lists the principal pressure-retaining materials and the appropriate material specifications for the RCPB components. DCD Tier 2, Revision 9, Table 6.1-1, lists the principal pressure-retaining materials and the appropriate material specifications of the containment system and the ECCSs.

DCD Tier 2, Revision 9, Section 6.1.1.2 states that all materials of construction used in essential portions of ESF systems are resistant to corrosion, both in the medium contained and the external environment.

DCD Tier 2, Revision 9, Section 6.1.1.2 also states that general corrosion of all materials, except carbon and low-alloy steel, is negligible and conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steel.

ESBWR core cooling water and containment sprays employ demineralized water with no additives, as stated in DCD Tier 2, Revision 9, Section 6.1.1.2. DCD Tier 2, Revision 9, Section 9.2.3, describes the water quality requirements. The applicant contends that leaching of chlorides from concrete and other substances is not significant and no detrimental effects occur on any of the ESF construction materials from allowable containment levels in the high-purity water. Thus, the applicant concludes that materials are compatible with the post-LOCA environment.

As described in DCD Tier 2, Revision 9, Section 6.1.1 the ESBWR design conforms to the guidance provided in the following:

- RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3.
- RG 1.36
- RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," Revision 1.
- RG 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973.
- Generic Letter (GL) 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping.”
- NUREG–0313, Revision 2, “Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.”

6.1.1.3 *Staff Evaluation*

6.1.1.3.1 Materials and Fabrication

To meet the requirements of GDC 1 and 10 CFR 50.55a to ensure that plant SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function they perform, the applicant must identify codes and standards and maintain records. Selection of the materials specified for use in these systems must be in accordance with the applicable provisions of Section III, Divisions 1 or 2, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, or RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 34. Section III references applicable portions of ASME Code, Section II, Parts A, B, C, and D.

DCD Tier 2, Revision 9, Table 6.1-1, lists the ASME Code classification and material specifications of components of the ESF systems. The staff reviewed the material specifications listed in Table 6.1-1 and verified that the aforementioned materials are acceptable for use in the ESBWR design in accordance with Section III of the ASME Code or RG 1.84. Given that DCD Tier 2, Section 6.1.1.1, states that Table 6.1-1 lists the principal pressure-retaining materials for the containment system and the ECCSs, the staff issued request for additional information (RAI) 6.1-1, asking the applicant to verify that all ESF materials meet the requirements of ASME Code, Section III, or the guidance of RG 1.84.

The applicant stated that materials for these systems must comply with American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, and therefore will only be materials that appear in ASME Code, Section III, Appendix I (now Section II, Part D), and that all such materials are in accordance with ASME Code, Section II, Parts A, B, or C, or RG 1.84. The applicant further stated that the design, fabrication, and testing requirements for ESF components, and fracture toughness requirements for all ferritic ESF materials in the ESBWR design will comply with the appropriate Section III class shown in DCD Tier 2, Section 6.1, Table 6.1-1.

In RAI 6.1-2, the staff asked the applicant to include weld filler metal specifications in Table 6.1-1. In response, the applicant provided filler metal specifications and classifications for weld filler metal used in the ESF systems with the exception of carbon steel and low-alloy steel filler materials. Given that the specifications for carbon and low-alloy steel listed by the applicant encompass a broad range of filler metal classifications, the staff considered this RAI response to be incomplete. In RAI 6.1-2 S01, the staff requested that the applicant include classifications of filler materials used to join carbon steel and low-alloy steel components in ESF systems. The applicant responded and proposed a revision to Table 6.1-1.

The applicant listed weld filler material classifications E9018-B3L and ER90S-B3L for use when welding low-alloy steel. The staff noted that ASME discontinued these weld filler material classifications and replaced them with classifications E8018-B3L and ER80S-B3L. DCD Tier 2, Revision 3, Table 5.2-4, contained similar inappropriate references to discontinued classifications. To determine that the weld filler materials used in the ESBWR design meet the requirements of ASME Code, Section II, Part C, “Specifications for Welding Rods, Electrodes, and Filler Metals.”, the staff issued RAI 6.1-2(a) S02, asking that the applicant modify DCD Tier 2, Tables 5.2-4 and 6.1-1 to include the correct weld filler material classifications.

The applicant’s proposed a revision to the weld filler material listed in DCD Tier 2, Table 6.1-1 that will be used to weld P5C, Group 1 (G1) materials. After reviewing the ESF material specifications provided by the applicant in the proposed revision to DCD Tier 2, Table 6.1-1, the staff was unable to identify any materials that fell into the P5C, G1 category in accordance with ASME Code, Section IX, Table QW-422, “Ferrous P-Numbers and S-Numbers.” To determine that the materials specifications and grades used in the ESBWR design met the requirements of ASME Code, Section II, Parts A “Ferrous Material Specifications,” B “Nonferrous Material Specifications,” and C, the staff issued RAI 6.1-2(b) S02, requesting that the applicant identify the P5C, G1 materials used in the ESBWR design for ESF components or else delete this information from the DCD if it does not apply. The staff noted that the same issue existed in DCD Tier 2, Revision 3, Table 5.2-4, in which the applicant referenced P5C, G1 materials as requiring welding, but the staff could not identify any P5C materials in the RCPB. Therefore, the staff also requested, as part of RAI 6.1-2(b) S02, that the applicant identify the P5C, G1 materials used in the ESBWR design for RCPB components or else delete this information from DCD Tier 2, Table 5.2-4 if it does not apply.

The applicant’s proposed revision to DCD Tier 2, Table 6.1-1 identified shielded manual arc welding filler material E8018-G for use in welding low-alloy steel in the ESBWR design. To complete its review and evaluate the applicant’s compliance with 10 CFR 50.55a, the staff issued RAI 6.1-2(c) S02, asking the applicant to provide the complete GE-Hitachi Nuclear Energy (GEH) specification that will be used to purchase E8018-G for fabricating ASME Code, Section III, Class 1, 2, and 3 components. In addition, the staff requested that the applicant provide a technical justification for using the GEH specification in lieu of commercially available welding electrodes. The staff identified the above issues regarding weld filler metal specifications and P numbers as RAI 6.1-2. RAI 6.1-2 was being tracked as an open item in the safety evaluation report (SER) with open items.

In response, the applicant indicated that it would modify Tables 6.1-1 and 5.2-4 to delete obsolete filler material classifications, delete references to P5C Group 1 materials, and delete E8018-G filler material classifications. The staff reviewed the ESBWR DCD Tier 2, Revision 5, and verified that the appropriate modifications were made. Based on the applicant’s response, RAI 6.1-2 is resolved.

The isolation condenser system (ICS) in the ESBWR design includes four isolation condensers (ICs), which are ASME Code, Section III, Class 2 components. In RAI 5.4-20, the staff requested that the applicant provide detailed information on the design of the ICs. In response to this RAI, the applicant indicated that the IC tubes would be fabricated from a modified form of Alloy 600 However, in other portions of its submittal, the applicant indicated that Alloy 600 would be used in the fabrication of the IC tubes. In RAI 5.4-20(D) the staff requested that the applicant clarify the material of construction for IC tubes. The applicant responded that the material of construction for the IC heat exchanger tubes will be modified SB-167 in accordance with Code Case N-580-1, "Use of Alloy 600 With Columbium Added Section III, Division 1." The staff confirmed that the applicant had appropriately modified DCD Tier 2, Table 6.1-1. RG 1.84 endorses Code Case N-580-1 for use, without conditions. The staff therefore finds this acceptable. Based on the applicant's response, RAI 5.4-20(D) regarding IC materials specifications is resolved.

As part of its response to RAI 5.4-20, the applicant indicated that the IC tubes will be bent by induction. However, the applicant did not indicate what effect, if any, this would have on the material properties of the tubing, nor did it indicate what testing, if any, was performed to confirm the acceptability of the material properties following bending of the piping/tubing. In RAI 5.4-20(A), the staff requested that the applicant discuss how it confirmed that the material properties of the most limiting bent tube remain acceptable following induction bending. The staff also requested that the applicant include a discussion of the material properties tested (e.g., hardness), the results, and the acceptance criteria. The applicant responded and indicated that although the hardware has not yet been fabricated, GEH will perform a qualification of induction-bent tubing. For tubes that will be subjected to induction bending after solution annealing, a qualification sample of the material will be subjected to mechanical testing (including yield, ultimate strength, and percent elongation). The acceptance criteria for this testing will be the mechanical properties listed in the material specification. Verification that testing is performed will be completed as part of DCD Tier 1, Revision 7, "ITAAC for The Isolation Condenser System," ITAAC 2a3, Table 2.4.1-3. The staff finds this acceptable because the applicant will provide a testing program for induction-bending operations that will ensure that the mechanical properties of the IC tubes required by the ASME Code will be acceptable following bending operations.

In RAI 5.4-20, the staff also requested that the applicant provide additional details on the design of the support structures for the IC tubes, if any, on the "pool side" and their materials of construction. RAI 5.4-20 was being tracked as an open item in the SER with open items. In response, the applicant indicated that the design of the support structures of the IC tubes is not currently available. The staff notes that, depending on the design, there may be crevices between the IC tube and the support. Such crevices could result in the accumulation of chemical contaminants that could lead to corrosion. In addition, the materials of construction of the support are important because any corrosion of them could result in a loss of support for, or damage to, the IC tubes. Given that material selection and specific design attributes, such as the presence of crevices, can contribute to degradation, the staff requested, in RAI 5.4-20(B) that the applicant provide a combined license (COL) information item to require submittal of this information. The applicant responded and stated that an ASME Code design specification, as well as a design report, will be available at the plant site for review. In addition, the applicant stated that crevices have been eliminated to the extent possible in the IC design. The applicant therefore believes that no COL information item is needed. The actual IC system operation will be less than 1,000 hours. The staff notes that the applicant indicated, in its response, that the normal operating temperature of the IC pool is less than 65 degrees Celsius (C) (149 degrees Fahrenheit [F]). Given that the normal operating temperature of the IC pool is relatively low, the amount of operating time is less than 1,000 hours, crevices have been eliminated to the extent possible in the IC design, and the IC pool is demineralized water with controlled impurity limits, the staff considers the likelihood of any significant degradation to be minimal. The staff therefore finds the applicant's decision not to include the aforementioned COL information item acceptable. Based on the applicant's response, RAI 5.4-20 is resolved.

In RAI 6.1-17, the staff requested that the applicant modify the containment liner materials listed in DCD Tier 2, Table 6.1-1 to be consistent with the liner materials listed in DCD Tier 2, Section 3.8. The applicant responded and modified Table 6.1-1 to reference DCD Tier 2, Section 3.8, for materials used for the containment vessel liner plate, penetrations, gravity-driven cooling system (GDCS) pool liner, and suppression pool liner. The staff reviewed the materials for the above components and verified that they are permitted for use in accordance with ASME Code, Section III, with the exception of American Society for Testing and Materials (ASTM) A709 "Standard Specification for Structural Steel for Bridges," Grade HPS 70W, which is not listed as a permitted material specification in accordance with ASME Code, Section III, Division II, Article CC-2000. The applicant indicated that it intends to use this material in accordance with ASME Code Case N-763, "ASTM A 709-06, Grade HPS 70W (HPS 485W) Plate Material Without Postweld Heat Treatment as Containment Liner Material or Structural Attachments to the Containment Liner, Subsection CC Section III, Division 2," for the containment liner and structural attachments welded to the containment liner. Code Case N-763 has gone through the ASME Committee approval process and has been found acceptable. ASTM A709 HPS 70W is a high-performance quenched and tempered weathering steel that is widely used in the fabrication of steel bridges. This material has high toughness in the as-welded condition and exhibits good resistance to corrosion when exposed to atmospheric conditions. The staff notes that ASTM A709 HPS 70W steel is currently permitted for use by American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities." Based on the above-listed considerations, the staff finds that the use of A709 HPS 70W is acceptable for its intended use. Based on the applicant's response, RAI 6.1-17 is resolved.

The staff finds that the ESF materials conform to ASME Code, Section III, and RG 1.84 and that the ESF materials meet the requirements of GDC 1 and 10 CFR 50.55a.

6.1.1.3.2 Austenitic Stainless Steels

The ESBWR design must meet the requirements of (1) GDC 4, relative to compatibility of components with their environmental conditions, (2) GDC 14, with respect to fabrication and testing of the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture, and (3) the QA requirements of Appendix B to 10 CFR Part 50. Designs may meet these requirements by following the guidance of GL 88-01; NUREG–0313, Revision 2; and RGs 1.31, 1.37, and 1.44. Designs must also provide controls over the use of cold-worked austenitic stainless steels.

For stainless steel components in the ESF systems, DCD Tier 2, Revision 9, Section 6.1.1.3, refers to DCD Tier 2, Revision 9, Section 5.2.3, for discussion of the fabrication and processing of austenitic stainless steels, as well as conformance to the regulatory guidance in RGs 1.31, 1.37, and 1.44; GL 88-01; and NUREG–0313, Revision 2. Section 5.2.3 of this report contains the staff's evaluation of the applicant's conformance to the aforementioned NRC documents. The staff has finds that the applicant either follows the guidance of, or has provided an acceptable alternative to, RGs 1.31, 1.37, and 1.44; GL 88-01; and NUREG–0313, Revision 2. The staff also finds that the applicant's controls over the use of cold-worked austenitic stainless steels, as discussed in DCD Tier 2, Revision 9, Sections 5.2.3 and 6.1.1.3.3, are acceptable because cold work will be controlled by the applicant during fabrication by applying limits in hardness, bend radii and the surface finish on ground surfaces which will reduce the susceptibility of components to stress-corrosion cracking.

6.1.1.3.3 Ferritic Steel Welding

To meet the requirements of GDC 1 related to general QA and codes and standards, Appendix B to 10 CFR Part 50 for control of special processes, and 10 CFR 50.55a, the amount of minimum specified preheat must meet ASME Code, Section III, Appendix D, Article D-1000, and RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," May 1973, unless an alternative procedure is justified. In addition, moisture control on low-hydrogen welding materials must conform to the requirements of ASME Code, Section III.

As requested by the staff, the applicant verified that minimum preheat requirements meet ASME Code, Section III, Appendix D, Article D-1000, and follow the guidelines of RG 1.50. For the standby liquid control (SLC) accumulator tank, the preheat recommendations of ASME Code, Section III, Appendix D, Article D-1000 will be followed. The applicant specified the use of an alternative to RG 1.50. The applicant's alternative consists of performing a postweld bakeout of welds that do not go directly from preheating temperature to postweld heat treatment. The staff concludes that the applicant's alternative to RG 1.50 is acceptable, given that it provides reasonable assurance that delayed hydrogen cracking will not occur between the completion of welding and postweld heat treatment. Section 5.2.3 of this report discusses the staff's evaluation of the applicant's alternative in more detail.

6.1.1.3.4 Dissimilar Metal Welds

The applicant described all dissimilar metal welds (DMWs) in the ESF systems and discussed the selection of filler metals, welding processes, and process controls for DMWs. The DMWs in the ESF will be performed with the same materials and process selections as the RCPB. In RAI 5.2-40 the staff reviewed the applicant's response and considers the applicant's description of its selection of filler metals, welding processes, and process controls acceptable, as they will provide reasonable assurance that the DMWs in the ESBWR design will maintain structural integrity throughout the design life of the plant. Section 5.2.3.3.1 of this report contains the staff's more detailed evaluation and resolution of this topic and RAI 5.2-40.

6.1.1.3.5 Limited Accessibility Welder Qualification

In RAI 6.1-6, the staff asked the applicant to verify that the ESBWR design related to fabrication of ESFs will follow the guidance in RG 1.71, "Welder Qualification for Areas of Limited Accessibility," Revision 1. The applicant responded that RG 1.71 will be applied to ESF systems in the same manner as for the RCPB systems. The staff finds the applicant's level of compliance with the guidelines detailed in RG 1.71 acceptable, as it will provide reasonable assurance that welds made under limited access conditions will be performed by personnel with appropriate qualifications to produce sound, high-quality welds. Section 5.2.3 of this report gives the staff's more detailed evaluation of the applicant's implementation of RG 1.71 for RCPB systems. The staff considers this RAI resolved.

6.1.1.3.6 Composition and Compatibility of ESF Fluids

The core cooling water and containment sprays in the ESBWR use demineralized water with no additives. The applicant indicated that materials used in essential portions of ESF systems are resistant to corrosion, both in the medium contained and the external environment. The applicant also stated that general corrosion of all materials, with the exception of carbon and low-alloy steels, is negligible and the ESBWR design provides conservative corrosion allowances for all exposed surfaces of carbon and low-alloy steel.

The process for determining the corrosion allowance for ferritic materials is the same as that applied to RCPB materials. The corrosion allowance is primarily based on GEH internal testing. The allowances consider fluid velocity, oxygen content, and temperature, and they include a safety margin over the actual measured corrosion rates of approximately a factor of 2. The designs of most operating boiling-water reactors (BWRs) (GEH design) have applied the same method, with corresponding allowances, including the certified advanced boiling-water reactor (ABWR) design. The staff considers the applicant's corrosion allowances acceptable, given that the ESBWR corrosion allowances for ferritic materials are based on laboratory testing, operational experience, and a safety margin of 2.

To meet the requirements of GDC 4, 14, and 41, the plant design should control the water used in the ESF to ensure against SCC in unstabilized stainless steel components. The staff reviewed the applicant's water quality requirements for the makeup water system demineralized water storage tank (DCD Tier 2, Revision 9, Table 9.2-7) and makeup water system demineralizer effluent (DCD Tier 2, Revision 9, Table 9.2-7). The chemistry control requirements of Tables 9.2-7 and 9.2-8 for conductivity, chloride, and pH in the ESBWR design are consistent within the limits listed in Section 6.1.1 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" March 2007 (hereafter referred to as the SRP), and are therefore acceptable.

DCD Tier 2, Revision 9, Table 6.1-1, indicates that Alloy 600 is used for IC tubing and header fabrication. Alloy 600 has a history of being susceptible to SCC in light-water reactor systems. In RAI 6.1-10, the staff asked the applicant to provide a basis for the use of Alloy 600 in the IC, including material condition (i.e., mill annealed or thermally treated) as it relates to susceptibility to SCC in the reactor coolant and demineralized water environment. In response, the applicant indicated that there have been no reports of Alloy 600 cracking in BWRs in the absence of a welded crevice or a crack initiated in adjacent Alloy 182. These initiating features are absent from the ESBWR design. In addition, the material used for the IC is the same alloy as used for reactor shroud support and stub tubes (see the response to RAI 4.5-18, as discussed in Section 4.5 of this report). This alloy (see ASME Code Case N-580-1) is a significantly modified version of Alloy 600, wherein the carbon content is limited, niobium (columbium) is added as a stabilizer, and high-temperature solution heat treatment is required instead of a mill anneal. Stress-corrosion resistance is very good. The alloy is approved for use by ASME Code Case N-580-1 and has been deployed in several operating BWRs, including the Kashiwazaki-Kariwa 6/7 ABWRs. Several of these units have been operating for more than 10 years. In RAI 5.4-55, the staff requested that the applicant discuss the corrosion allowances for Alloy 600 used in the ICs. RAI 5.4-55 was being tracked as an open item in the SER with open items. In response, the applicant indicated that the Alloy 600 tubing in early boiling-water reactor (BWR) ICs performed satisfactorily, with no incidents resulting from general corrosion in this application. Although general corrosion is a concern, the applicant did not address whether any other incidences of corrosion or other degradation have occurred in operating units. In RAI 5.4-55 S01, the staff requested that the applicant discuss whether there have been any other "incidents" associated with the use of these materials in these applications. The applicant responded and indicated that a review of IC industry experience did not identify any incidents associated with the use of Alloy 600 material. Based on the applicant's response, RAIs 5.4-55 and 6.1-10 are resolved.

6.1.1.3.7 Component and Systems Cleaning

The staff reviewed the ESF structural materials to ensure that the requirements of Appendix B to 10 CFR Part 50 were met, as they relate to the establishment of measures to control the cleaning of material and equipment. The controls established for cleaning of material and equipment must be performed in accordance with work and inspection instructions to prevent damage or deterioration.

The ESBWR design complies with RG 1.37, except as noted in DCD Tier 2, Revision 9, Table 1.9-21B. Table 2-1 of NEDO-11209-04a, Revision 8, “GE Nuclear Energy Quality Assurance Program Description,” Class I (nonproprietary), documents the alternative that the applicant may use. The alternative involves using methods, other than mechanical ones, to remove local rusting on corrosion-resistant alloys. The NRC approved this alternative on March 31, 1989. Therefore, the applicant’s request to use this alternative is acceptable. Section 4.5.1.2.5 of this report further discusses the applicant’s level of compliance with RG 1.37. Thus, the ESBWR design satisfies the QA requirements of Appendix B to 10 CFR Part 50 for component and system cleaning.

6.1.1.3.8 Thermal Insulation

The type of thermal insulation used in the ESBWR containment will be primarily metallic and metal-encapsulated insulation. In DCD Tier 2, Revision 9, Section 6.1.1.3.4, the applicant stated that nonmetallic thermal insulation materials used on ESF systems are selected, procured, tested, and stored in accordance with RG 1.36.

To meet the requirements of GDC 1, 14, and 31, ESF systems should be designed, fabricated, erected, and tested such that there is an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The levels of leachable contaminants in nonmetallic insulation materials that come into contact with 300 series austenitic stainless steels used in fluid systems important to safety should be under careful control so as not to promote SCC. In particular, the leachable chlorides and fluorides should be held to the lowest levels practical. The staff’s position is that following the guidance in RG 1.36 is an acceptable method to control leachable contaminants in nonmetallic insulation materials. The applicant has stated that it will follow the guidance in RG 1.36, and the staff finds this acceptable as it will meet the requirements of GDC 1, 14, and 31.

6.1.1.4 *Conclusions*

Based on its review of the information provided by GEH, the staff concludes that the ESBWR DCD specifications for the materials to be used in the fabrication of the ESFs are acceptable and meet the relevant requirements of GDC 1, 4, 14, 31, 35, and 41; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a.

6.1.2 Organic Materials

6.1.2.1 *Regulatory Criteria*

The staff reviewed the protective coating systems (paints) and organic materials in accordance with SRP Section 6.1.2, Revision 3. Staff acceptance is based on meeting the requirements of Appendix B to 10 CFR Part 50 as it relates to the QA requirements for the design, fabrication, and construction of safety-related SSCs. To meet the requirements of Appendix B to

10 CFR Part 50, the applicant can specify that the coating systems and their applications will follow the guidance of RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 1.

6.1.2.2 *Summary of Technical Information*

The ESBWR design has reduced the use of coatings inside containment to a minimum. The areas in which most of the coatings are used are the following:

- Internal steel structures
- Carbon steel containment liner
- Equipment inside drywell and wetwell

DCD Tier 2, Revision 9, states that all field-applied epoxy coatings inside containment will meet the requirements of RG 1.54 and are qualified using the standard ASTM tests, as applicable to procurement, installation, and maintenance.

6.1.2.3 *Staff Evaluation*

The staff reviewed the protective coating systems (paints) and organic materials in accordance with SRP Section 6.1.2, Revision 3. Staff acceptance is based on meeting the requirements of Appendix B to 10 CFR Part 50, as it relates to the QA requirements for the design, fabrication, and construction of safety-related SSCs. To meet the requirements of Appendix B to 10 CFR Part 50, the applicant should specify that the coating systems and their applications will follow the guidance of RG 1.54, Revision 1. This RG references the QA standards of ASTM D3842, “Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants”; ASTM D3911, “Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions”; and ASTM D5144-00, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants.”

RG 1.54, Revision 1, provides guidance on practices and programs that are acceptable to the staff for the selection, application, qualification, inspection, and maintenance of protective coatings applied in nuclear power plants. In addition, this latest revision to the RG updates the definitions of Service Level I, II, and III coating locations to include both safety-related and nonsafety-related regions, as set forth by the ASTM Committee and the updated ASTM guidance.

The applicant stated that the protective coating system meets the regulatory positions of RG 1.54, Revision 1, and the standards of ASTM D5144-00, as applicable.

The applicant also stated that not all coatings inside containment will meet the criteria of RG 1.54, Revision 1, and ASTM D5144-00. The exceptions are for small equipment where, in case of a LOCA, paint debris is not a safety hazard. To address this issue, the applicant included a commitment that the COL applicant is required to do the following:

- Describe the approach to be taken to identify and quantify all organic materials that exist within the containment building in significant amounts that do not meet the requirements of ASTM D5144-00 and RG 1.54, Revision 1, as per SRP Section 6.1.2.

- Provide the milestone when evaluations will be complete to determine the generation rate, as a function of time, of combustible gases that can be formed from these unqualified organic materials under design-basis accident (DBA) conditions.

- As part of these evaluations, provide the technical basis and assumptions used.

This was identified as COL Information Item 6.1-1-A (subsequently deleted) in DCD Tier 2, Revision 3, Section 6.1.3.1.

Because the amount of organic materials does not meet the requirements of RG 1.54 and will not be available before the procurement of the components, the staff requested, in RAI 6.1-16, that the applicant revise the DCD (including addressing a COL information item) to ensure that the COL applicant provides a bounding value for the amount of unqualified coatings and the assumptions used to determine this bounding value. In Revision 5 of DCD Tier 1, the applicant deleted COL Information Item 6.1-1-A and revised the DCD to specify that all field-applied epoxy coatings inside containment will meet the requirements of RG 1.54 and that the coatings are qualified using the standard ASTM tests. In addition, consistent with the rationale of RG 1.54, the wetwell and attendant vertical vents are designated as a Service Level I area. All surfaces and equipment in this area are either uncoated, corrosion-resistant stainless steel, or coated in accordance with RG 1.54 and referenced ASTM standards, as applicable. The staff finds Revision 5 of the DCD acceptable because all field-applied epoxy coatings inside containment will meet the requirements of RG 1.54 and the coatings are qualified using the standard ASTM tests. Based on the applicant's response, RAI 6.1-16 is resolved.

6.1.2.4 *Conclusions*

The staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on the applicant having met the QA requirements of Appendix B to 10 CFR Part 50, as the coating systems and their applications will meet the requirements of RG 1.54, Revision 1. By meeting the recommendations in RG 1.54, Revision 1, the COL applicant will have evaluated the suitability of the coatings to withstand a postulated DBA environment, in accordance with NRC accepted practices and procedures.

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 *Pressure Suppression Containment*

6.2.1.1.1 Regulatory Criteria

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 6.2.1.1, in accordance with SRP Section 6.2.1, Revision 3, issued March 2007; SRP Section 6.2.1.1.C, Revision 7, issued March 2007; and SRP Section 6.2.1.3, Revision 3, issued March 2007.

In accordance with SRP Section 6.2.1.1.C, Revision 7, acceptance criteria are based on the following GDC, which apply to the design and functional capability of a BWR pressure-suppression type containment:

- GDC 4 requires that SSCs important to safety be designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during normal plant operation or following a LOCA.

- GDC 16, “Containment design,” and GDC 50, “Containment design basis,” as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- GDC 53, “Provisions for containment testing and inspection,” as it relates to (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

To meet the requirements of GDC 16 and 50 regarding the design margin for the ESBWR, which is similar in design to a BWR III plant, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values. To meet the requirement of GDC 16, provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) against loss of integrity from negative pressure transients or post accident atmosphere cooldown:

- Structures should be designed to withstand the maximum calculated external pressure.

- Vacuum relief devices should be provided in accordance with the requirements of the ASME Code, Section III, Subsection NE, to ensure that the external design pressures of the structures are not exceeded.

The maximum allowable leakage area for steam bypass of the suppression pool should be greater than the technical specification (TS) limit for leakage measured in periodic drywell-wetwell leakage tests to demonstrate that the design meets the requirement of GDC 53 regarding periodic testing at containment design pressure.

6.2.1.1.2 Summary of Technical Information

The containment systems for the ESBWR include a containment structure and a reactor building (RB) surrounding the containment structure and housing equipment essential to safe shutdown of the reactor. The containment is designed to prevent the uncontrolled release of radioactivity to the environment with a leakage rate of 0.35 percent by weight per day at the calculated peak containment pressure related to the DBA. The RB is designed to provide an added barrier to the leakage of airborne radioactive materials from the primary containment in case of an accident. ESBWR DCD Tier 2, Figure 6.2.1, shows the principal features of the ESBWR containment.

The ESBWR containment is designed with the following main features:

- The drywell consists of (1) an upper drywell volume surrounding the upper portion of the reactor pressure vessel (RPV) and housing the main steam and feedwater piping, GDCS pools and piping, passive containment cooling system (PCCS piping, ICS piping, safety/relief valves (SRVs) and piping, depressurization valves (DPVs) and piping, drywell coolers and piping, and other miscellaneous systems, and (2) a lower drywell volume below the RPV support structure housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems and equipment below the RPV, and vessel bottom drain piping.

- The upper drywell is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV support structure separates the lower drywell from the upper drywell. There is an open communication path between the two drywell volumes via upper drywell to lower drywell connecting vents, built into the RPV support structure. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping, and electrical and instrumentation lines are provided with seals and leaktight connections.

- The drywell, which has a net free volume of 7,206 cubic meters (m^3) (254,500 cubic feet [ft^3]), is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the drywell and also the negative differential pressures associated with containment depressurization events, when the steam in the drywell is condensed by the PCCS, the GDCS, the fuel and auxiliary pools cooling system (FAPCS), and cold water cascading from the break following post-LOCA flooding of the RPV. The drywell design pressure and temperature are 310 kilopascals gauge (kPaG) (45 pounds per square inch gauge [psig]) and 171 degrees C (340 degrees F), respectively. The design drywell minus wetwell differential pressure is 241 kilopascals differential (kPaD) (35 pounds per square inch differential [psid]) to -20.7 kPaD (-3.0 psid). The design drywell internal minus external differential pressure is -20.7 kPaD (-3.0 psid).

- The wetwell consists of a gas volume and a suppression pool, with a net gas volume of 5,350 m^3 (188,900 ft^3) and a normal pool volume of 4,424 m^3 (156,200 ft^3) at low water level.

- The wetwell is designed for an internal pressure of 310 kPaG (45 psig) and a temperature of 121 degrees C (250 degrees F).

- The suppression pool, which is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water that will absorb energy by condensing steam from safety relief valve (SRV) discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The flow path to the wetwell is designed to entrain radioactive materials by routing fluids through the suppression pool during and following a LOCA. The gas space above the suppression pool is leaktight and sized to collect and retain the drywell gases following a pipe break in the drywell, without exceeding the containment design pressure.

- Following a postulated DBA, the mass and energy released to the drywell will be transferred to the wetwell through a system of 12 vertical circular channels of a nominal diameter of 1.2 meters (m) (3.9 feet [ft]), each containing 3 horizontal vents of a nominal diameter of 0.70 m (2.3 ft), for a total of 36 vents. The three-vent centerlines in each column are located at 1.95 m (6.4 ft), 3.32 m (10.9 ft), and 4.69 m (15.4 ft) below the suppression pool water level when the suppression pool is at the low water level.

- A spillover system provides drywell to wetwell connection to limit suppression pool drawdown and the holdup volume in the drywell following a LOCA by transferring water from the drywell annulus to the suppression pool. Spillover is accomplished by 12 horizontal holes (200-millimeter [mm] [7.87 inch [in.]] nominal diameter), which are built into the vent wall connecting the drywell annulus with each vertical vent module. If water ascending through the drywell annulus following a postulated LOCA reaches the spillover holes, it will flow into the suppression pool via the vertical/horizontal vent modules. Once in the suppression pool, the water can be used for accident mitigation (i.e., by restoration of RPV inventory).

- A drywell-to-wetwell vacuum breaker system protects the integrity of the diaphragm floor slab and vent wall between the drywell and the wetwell, and the drywell structure and liner, and will prevent back-flooding of the suppression pool water into the drywell. The vacuum breaker is a process-actuated valve, similar to a check valve, and is provided with redundant proximity sensors to detect its closed position. On the upstream side of each vacuum breaker, pneumatically operated fail-as-is safety-related isolation valves are provided to isolate a leaking (not fully closed) or stuck open vacuum breaker. During a LOCA, the vacuum breaker opens and allows the flow of gas from wetwell to drywell to equalize the drywell and wetwell pressure. After the drywell and wetwell pressure equalizes, the vacuum breaker closes to prevent extra bypass leakage caused by the opening created by the vacuum breaker, and, therefore, to maintain the pressure suppression capability of the containment. If the vacuum breaker does not completely close, as detected by the proximity sensors, a control signal will close the upstream backup valve. Redundant vacuum breaker systems are provided to protect against a single failure of a vacuum breaker, either failure to open or failure to close when required.

Similar to an ABWR, the ESBWR containment design uses combined features of the Mark II and Mark III designs, except that the drywell consists of upper drywell and lower drywell volumes.

The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark III wetwell.

<u>Vacuum Breakers</u>. Vacuum breakers are provided between the drywell and wetwell. The vacuum breaker is a self-actuating valve, similar to a check valve. The purpose of the drywell-to-wetwell vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the drywell and the wetwell, and the drywell structure and liner, and to prevent back-flooding of the suppression pool water into the drywell. The vacuum breaker is provided with redundant proximity sensors to detect its closed position. One out of the three vacuum breakers is required to perform the vacuum relief function. The third vacuum breaker provides redundancy, while the second vacuum breaker provides single-failure protection for opening. On the upstream side of each vacuum breaker, a pneumatically operated fail-as-is safety-related isolation valve is provided to isolate a leaking or stuck-open vacuum breaker. During a LOCA, the vacuum breaker opens and allows the flow of gas from wetwell to drywell to equalize the drywell and wetwell pressure. After the drywell and wetwell pressure equalizes, the vacuum breaker closes to prevent extra bypass leakage caused by the opening created by the vacuum breaker, and therefore, to maintain the pressure suppression capability of the containment. If the vacuum breaker does not completely close, as detected by the proximity sensors, a control signal will close the upstream backup valve.

Redundant vacuum breaker systems are provided to protect against a single failure of a vacuum breaker, either failure to open or failure to close when required. DCD Tier 2, Revision 9, Table 6.2-1 provides the design drywell-to-wetwell pressure difference and the vacuum breaker full-open differential pressure.

The vacuum breaker valves are protected from pressure suppression loads by structural shielding designed for pressure suppression loads based on a Mark II/III containment design.

<u>Steam Bypass of the Suppression Pool</u>. The pressure suppression containment is designed such that any steam released from a pipe rupture in the primary system is condensed by the suppression pool and does not produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. If a leakage path were to exist between the drywell and the suppression pool (wetwell) gas space, the leaking steam would produce undesirable pressurization of the containment. The bounding DBA calculation assumes a bypass leakage area (expressed as the leak flow area divided by the square root of the leak K-loss coefficient [$A/\sqrt{K}$]) of 2 square centimeters (cm^2) (0.31 $in.^2$) as specified in TS Surveillance Requirement (SR) 3.6.2.2.2. In the ESBWR design, the PCCS also condenses some of the steam released from the pipe rupture.

<u>Loss-of-Coolant Accidents</u>. The staff based its containment functional evaluation on the GEH consideration of a representative spectrum of postulated LOCAs, which would result in the release of reactor coolant to the containment. These LOCAs include the following:

- Liquid line breaks
  - An instantaneous guillotine rupture of a feedwater line (FWL)
  - An instantaneous guillotine rupture of a GDCS line
  - An instantaneous guillotine rupture of a vessel bottom drain line

- Steamline breaks
  - An instantaneous guillotine rupture of a main steamline (MSL)

GEH used the TRACG computer program to evaluate the containment performance, as described in NEDC-33083P-A, “TRACG Application for ESBWR,” issued March 2005, and NEDE-32176P, “TRACG Model Description,” issued January 2008. The staff’s safety evaluation in Section 4 of NEDC-33083P-A contains items needing confirmation during the ESBWR design certification stage. The staff addresses these confirmatory items in the “Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design.”

DCD Tier 2, Revision 9, Tables 6.2-1 through 6.2-4, list key design and operating parameters of the containment system, including the design characteristics of the drywell, the wetwell, and the pressure-suppression vent system and key assumptions used for the DBA analysis. DCD Tier 2, Revision 9, Tables 6.3-1 through 6.3-3 provide the performance parameters of the related emergency safety feature systems, which supplement the design conditions of DCD Tier 2, Revision 9, Table 6.2-1, for containment performance evaluation. DCD Tier 2, Revision 9, Table 6.2-6, provides the nominal and bounding values for the plant initial and operating conditions for evaluating the containment performance.

Using the nominal initial and operating conditions listed in DCD Tier 2, Table 6.2-1, GEH evaluated four cases, the three liquid line break cases and the steamline break case. The results of the four cases showed that instantaneous guillotine ruptures of an MSL and an FWL gave the highest containment pressure. GEH then used the bounding initial and operating conditions listed in Table 6.2-1 in its evaluation of the main steamline break (MSLB) and the feedwater line break (FWLB) cases. Results of these analyses show that an instantaneous guillotine rupture of an MSL with failure of one DPV produced the most limiting responses for the containment pressure evaluation. The second limiting case is an instantaneous guillotine rupture of an FWL with failure of one SRV. DCD Tier 2, Revision 9, Table 6.2-5, lists the results of GEH evaluations of the four cases using the nominal initial and operating conditions and the five cases using bounding initial and operating conditions.

Negative Pressure Design Evaluation. During normal plant operation, the inerted wetwell and the drywell volumes remain at a pressure slightly above atmospheric conditions. Certain events could lead to a depressurization transient that can produce a negative pressure differential in the containment. A drywell depressurization results in a negative pressure differential across the drywell walls, vent wall, and diaphragm floor. A negative pressure differential across the drywell and wetwell walls means that the RB pressure is greater than the drywell and wetwell pressures, and a negative pressure differential across the diaphragm floor and vent wall means that the wetwell pressure is greater than the drywell pressure. If not mitigated, the negative pressure differential can damage the containment steel liner. The ESBWR design provides the vacuum relief function necessary to limit these negative pressure differentials to within design values.

The following events may cause containment depressurization:

- Post-LOCA drywell depressurization is caused by the ECCS (e.g., GDCS, control rod drive [CRD] system) flooding of the RPV and cold water spilling out of the broken pipe or cold water spilling out of the broken GDCS line directly into the drywell.

- The drywell sprays are inadvertently actuated during normal operation or during the post-LOCA recovery period.

- The combined heat removal of the ICS and PCCS exceeds the rate of decay heat steam production.

GEH expects drywell depressurization following a LOCA to produce the most severe negative pressure transient condition in the drywell. The results of the MSLB analysis show that the containment did not reach negative pressure relative to the RB and the maximum wetwell-drywell differential pressure was within the design capability. This calculation assumed one available vacuum breaker with an area of 9.67×10^{-2} square meters (m^2) (1.041 square feet [ft^2]). The calculation also assumed a drywell spray flow rate of 127 m^3/hour (h) (560 gallons per minute [gpm]) at a temperature of 293 Kelvin (67.7 degrees F) which is conservatively initiated when the drywell pressure has peaked just before opening of the vacuum breakers.

6.2.1.1.3 Staff Evaluation

For pressure-suppression type BWR plant containments, the staff review covers the following areas:

- The temperature and pressure conditions in the drywell and wetwell that result from a spectrum (including break size and location) of postulated LOCAs

- Suppression pool dynamic effects during a LOCA or following the actuation of one or more reactor coolant system SRVs, including vent clearing, vent interactions, pool swell (PS), pool stratification, and dynamic symmetrical and asymmetrical loads on suppression pool and other containment structures

- The consequences of a LOCA occurring within the containment (wetwell or outside the drywell)

- The capability of the containment to withstand the effects of steam bypassing the suppression pool

- The external pressure capability of the drywell and wetwell and systems that may be provided to limit external pressures

- The effectiveness of static and active heat removal mechanisms

- The pressure conditions within subcompartments and acting on system components and supports as a result of high-energy line breaks (HELBs)

- The range and accuracy of instrumentation provided to monitor and record containment conditions during and following an accident

- The suppression pool temperature limit during reactor coolant system SRV operation, including the events considered in analyzing suppression pool temperature response, assumptions used for the analyses, and the suppression pool temperature monitoring system

- The reactor coolant system SRV in-plant confirmatory test program

- The evaluation of analytical models used for containment analysis

DCD Tier 2, Revision 4, does not describe a chronology of progression of a LOCA, how it affects the containment and its systems, or how containment systems operate to mitigate the consequences of a LOCA. In RAI 6.2-175, the staff requested that GEH add this information to the DCD. RAI 6.2-175 was being tracked as an open item in the SER with open items. In response to RAI 6.2-175, GEH added Appendix E to DCD Tier 2, Revision 5, to provide the chronology of progression of a LOCA as predicted by TRACG containment analysis. This addressed the staff's concern. The staff's evaluation of TRACG LOCA containment analysis and staff's confirmatory analysis are described later in this section. RAI 6.2-175 is resolved.

Table 6.2-1 of this report reproduces DCD Tier 2, Table 6.2-6. DCD Tier 2, Revision 4, Table 6.2-6, listed the RPV nominal water level as "NWL." However, NWL was not defined in the Global Abbreviations and Acronyms List, and its value was not given in DCD Tier 2. In RAI 6.2-174, the staff asked GEH to define NWL and provide its value. In its response, GEH defined NWL as "normal water level" and added a footnote to DCD Tier 2, Revision 5, Table 6.2-6, stating that the NWL value is provided in DCD Tier 2, Revision 5, Table 15.2-1. RAI 6.2-174 was being tracked as an open item. The staff confirmed that this information was incorporated in DCD Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-174 is resolved.

Table 6.2-1 of this document shows the major plant initial and operational parameters used in the containment analysis.

Table 6.2-1. Plant Initial and Operating Conditions Considered in the Containment Performance Evaluation Cases.

No.	Plant Parameter	Nominal Value	Bounding Value
1	RPV Power	100%	102%
2	Wetwell relative humidity	100%	100%
3	PCC pool level	4.8 m (15.8 ft)	4.8 m (15.8 ft)
4	PCC pool temperature	43.3 °C (110 °F)	43.3 °C (110 °F)
5	Drywell pressure	101.3 kPa (14.7 psia)	106.9 kPa (15.5 psia)
6	Drywell temperature	46.1 °C (115 °F)	46.1 °C (115 °F)
7	Wetwell pressure	101.3 kPa (14.7 psia)	106.9 kPa (15.5 psia)
8	Wetwell temperature	43.3 °C (110 °F)	43.3 °C (110 °F)
9	Suppression pool temperature	43.3 °C (110 °F)	43.3 °C (110 °F)
10	GDCS pool temperature	46.1 °C (115 °F)	46.1 °C (115 °F)
11	Suppression pool level	5.45 m (17.9 ft)	5.50 m (18.1 ft)
12	GDCS pool level	6.60 m (21.7 ft)	6.60 m (21.7 ft)
13	Drywell relative humidity	20%	20%
14	RPV pressure	7.17 MPa (1040 psia)	7.274 MPa (1055 psia)
15	RPV water level	NWL*	NWL* + 0.3 m (1 ft)
16	RPV Dome Vapor and Saturation Temperature	287.4°C (549.3°F)	288.4°C (551.0°F)
17	RPV Lower Plenum Liquid Temperature	272.3°C (522.2°F)	272.2°C (522.0°F)

* NWL—Normal Water Level, 20.72 m (815.7 in.)

Vacuum Breakers. Section B.3.b of Appendix A to SRP Section 6.2.1.1.C specifies that the operability of all vacuum valves should be tested at monthly intervals to ensure free movement of the valves. Operability tests are conducted at plants of earlier BWR designs using an air-actuated cylinder attached to the valve disk. The air-actuated cylinders have proven to be one of the root causes of vacuum breakers failing to close. Free movement of the vacuum breakers in the ESBWR design has been enhanced by eliminating this potential actuator failure mode, improving the valve hinge design, and selecting materials that are resistant to wear and galling. Therefore, GEH considers this requirement for monthly testing unnecessary for the ESBWR. However, the vacuum breakers will be tested for free movement during each outage. The operability of the vacuum beakers is verified according to TS 3.6.1.6, “Suppression Wetwell-to-Drywell Vacuum Breakers.”

The staff finds that testing ESBWR vacuum breakers during each outage is acceptable for several reasons. First, proximity sensors are provided to detect if a vacuum breaker is not fully closed. Second, on the upstream side of each vacuum breaker, a pneumatically operated fail-as-is safety-related isolation valve is provided. Third, the containment analysis assumed that only two of three vacuum breakers would operate following a LOCA, thereby providing a level of redundancy to address potential failure of a vacuum breaker (DCD Tier 2, Revision 9, Section 6.2.1.1.3.1).

ESBWR DCD Tier 2, Revision 3, did not provide the vacuum breaker opening and closing differential pressure settings used in the TRACG containment analysis of the DBA. Therefore, in RAI 6.2-99, the staff asked GEH to provide this information. In response, GEH provided the information, but it was also necessary that the information be added to the DCD. RAI 6.2-99 was being tracked as an open item in the SER with open items. The staff confirmed that the information was incorporated in DCD Tier 2, Revision 4, Table 6.2-1, which addressed the staff's concern. Based on the applicant's response, RAI 6.2-99 is resolved.

In response to RAI 6.2-59, GEH stated that "[t]he ESBWR design uses 3 vacuum breakers. Assuming one vacuum breaker is out of service for the LOCA analyses, there should be 2 vacuum breakers available for the LOCA transient." Making three vacuum breakers available during a LOCA appears to be more conservative, considering that a higher rate of noncondensable gas flow from the wetwell to drywell would degrade the PCCS more than when only two vacuum breakers are available. Therefore, in RAI 6.2-142, the staff requested that GEH explain this apparent nonconservative modeling of only two of three vacuum breakers being available during a LOCA. In response, GEH stated that vacuum breakers open during the early phase of the transient, and the maximum containment pressure for the period of 72 hours following a LOCA occurs at the end of this period. Therefore, having two versus three vacuum breakers open was expected to have a minimal impact on the PCCS performance in the long term and thus on the maximum containment pressure. The applicant's response addresses the staff's concern and is acceptable because the applicant correctly described the effect of two versus three vacuum breakers opening. Based on the applicant's response, RAIs 6.2-142 and 6.2-59 are resolved.

Steam Bypass of the Suppression Pool. The potential exists for steam to bypass the suppression pool by various leak paths, primarily through the vacuum breakers. In response to RAI 6.2-12, GEH stated that a sensitivity analysis showed that the peak drywell pressure of an FWLB accident would approach the design pressure of 310 kPaG (45 psig) at 72 hours after the pipe break, if the leakage size were increased to $(A/\sqrt{K}) = 100\ cm^2$ ($0.107\ ft^2$). In RAI 6.2-147, the staff asked GEH to add this information to the DCD. In response, GEH stated that the containment analysis results included in DCD Tier 2, Revision 3, Section 6.2, indicate that the bounding LOCA break is an MSLB instead of an FWLB as reported in DCD Tier 2, Revision 2, Section 6.2. GEH referred to the containment analysis of an MSLB described in DCD Tier 2, Revision 3, Section 6.2.1.1.5.1, which states that the containment pressure remains below the design capability of the drywell with a bypass leakage of $2\ cm^2$ ($2.16\times10^{-3}\ ft^2$) $(A/\sqrt{K})$. Therefore, the bypass leakage of $100\ cm^2$ ($0.107\ ft^2$) $(A/\sqrt{K})$ is no longer limiting, and a DCD update is not needed. The applicant's response addresses the staff's concern and is acceptable because the staff's confirmatory analysis confirms the applicant's conclusions in Appendix E to DCD Tier 2. Based on the applicant's response, RAIs 6.2-147 and 6.2-12 are resolved.

DCD Tier 2, Revision 2, Section 6.2.1.1.5.1, states that the bounding design-basis calculation assumed a bypass leakage of $1\ cm^2$ ($0.001\ ft^2$) $(A/\sqrt{K})$. This value is significantly lower than the design capacities of Mark I, II, and III containments, which are 18.6, 46.5, and 929 cm^2 (0.02, 0.05, and 1.0 ft^2) ($A/\sqrt{K}$), respectively (SRP Section 6.2.1.1.C, Revision 6).

DCD Tier 2, Revision 2, Section 6.2.1.1.5.4.3, states that the acceptance criterion for the bypass leakage area for the leakage tests will be 10 percent of 1 cm^2 (0.001 ft^2) ($A/\sqrt{K}$) (i.e., 0.1 cm^2 (1×10^{-4} ft^2) [$A/\sqrt{K}$]). The staff was concerned that this may be a low value for bypass leakage, which plants may find difficult to confirm. Therefore, in RAI 6.2-145, the staff asked GEH to verify that plants will be able to measure such a low bypass leakage value.

In response, GEH proposed an alternative acceptance criterion for the bypass leakage area for the leakage tests—the leakage which is analytically required to keep the containment below design pressure, 2 cm^2 (2.16×10^{-3} ft^2) ($A/\sqrt{K}$). GEH argued that the ability of the containment to tolerate degraded (increased) leakage up to ultimate strength had been determined to be more than a factor of 5 above the design capability. In RAI 6.2-145 S01, the staff stated its position that the containment design pressure, but not the containment ultimate pressure, should be used for determining design margins. The staff stated that the GEH proposed bypass leakage criterion was unacceptable and requested that GEH propose an acceptable bypass leakage acceptance criterion. RAI 6.2-145 was being tracked as an open item in the SER with open items.

In response, GEH proposed (1) to increase the acceptance criterion for the suppression pool bypass leakage test to a value less than or equal to 1 cm^2 (1.08×10^{-3} ft^2) ($A/\sqrt{K}$), which amounts to 50 percent of the design-basis bypass leakage value, and (2) to increase the frequency of the overall suppression pool bypass leakage test to be the same as the integrated leak rate test (ILRT) frequency. GEH stated that General Electric established 10 percent of the containment capacity as the acceptance criterion for the suppression pool bypass leakage test during licensing of the initial pressure suppression containments in the early 1970s for BWRs with an active ECCS. GEH stated that the value of 10 percent of containment capability was intended to leave sufficient margin for increases in bypass leakage between outages, and it was chosen, in part, because of the limited amount of field-testing experience and data and the large number of penetrations through the diaphragm floor of the Mark II containment. In support of its position, GEH provided bypass leakage test data for Mark II containments.

These data show that, for each plant, the measured bypass leakages are significantly less than the surveillance test acceptance criteria. These data also show that plants have measured significantly lower bypass leakages than the leakage proposed for the ESBWR. In addition, each ESBWR vacuum breaker consists of an upstream isolation valve, which can isolate a leaking vacuum breaker during a LOCA upon detecting the leakage. Vacuum breakers are equipped with temperature gauges for detecting a leakage. Therefore, the staff finds that the bypass leakage surveillance criterion of 50 percent of the design value proposed is acceptable for the ESBWR.

When proposing in its response to increase the overall suppression pool bypass leakage test frequency to the same frequency as the ILRT, GEH stated that this frequency was similar to that employed at the following operating BWRs with Mark II containments: Columbia Generating Station, Nine Mile Point Unit 2, Susquehanna Units 1 and 2, and Limerick Units 1 and 2. Since the extensions to test frequency for the above plants were approved based on plant-specific data, the staff requested in RAI 6.2-145 that GEH provide additional justification for the proposed change for the ESBWR. Instead, in response, GEH changed the overall suppression pool bypass leakage test frequency to once every 24 months and made appropriate changes to the DCD.

RAI 6.2-145 was being tracked as an open item. The applicant's response is acceptable because the staff agrees with the applicant's rationale for the 24-month bypass leakage test frequency. Based on the applicant's response, RAI 6.2-145 is resolved.

DCD Tier 2, Revision 2, Section 6.2.1.1.2 states that "[o]n the upstream side of the vacuum breaker, a DC solenoid operated isolation valve designed to fail-close is provided." The vacuum breaker isolation valve (VBIV) provides a safety function of closing a leaking vacuum breaker. A vacuum breaker leaking at a rate higher than its design leakage value would cause steam to leak from the drywell to the wetwell bypassing the suppression pool at a rate higher than the design steam leakage value. Steam that enters the wetwell bypassing the suppression does not get condensed by the suppression pool and raises the wetwell pressure and eventually the drywell pressure. In RAI 6.2-148 staff asked GEH to state the type of isolation valve and how the fail-close function is provided.

In response GEH stated the following:

> VBIV is a pneumatically operated fail-as-is safety-related valve that isolates a leaking or stuck open vacuum breaker. Both the vacuum breaker and VBIV are located in the drywell side of the diaphragm floor. The VBIV valve type will be of similar design to a triple offset metal-seated butterfly valve. Automatic actuation logic will close the VBIV based upon an open indication provided by the vacuum breaker proximity sensors with temperature confirmation or indication of bypass leakage provided by temperature sensors. These temperature sensors are located within the cavity of the vacuum breaker/VBIV assembly. Additional temperature sensors are located in close proximity to the vacuum breaker outlets screens and in the drywell and wetwell.

GEH stated that during a LOCA, if a vacuum breaker leaks, these same temperature sensors will detect a decrease in temperature differential between the hot drywell gas leaking past the vacuum breaker seat and the wetwell gas. This will generate a signal to close the VBIV. Proximity sensors located on the vacuum breaker seat can also generate a close signal if they detect a stuck-open vacuum breaker coincident with a separate temperature confirmation.

The GEH response did not provide information on the limit of bypass leakage that activates the sensors to close the VBIV and the value of temperature differential that activates the sensors. Therefore, in RAI 6.2-148 S01, the staff asked GEH to provide this information.

In response GEH stated that a vacuum breaker not fully closing, which is considered a single failure, is defined as a bypass leakage area greater than 0.6 cm^2 (0.093 in^2) ($A/\sqrt{K}$). GEH stated that "DCD, Tier 1, Table 2.15.1-2, ITAAC 16b will be changed to a type test to detect bypass leakage from 0.3 cm^2 to 0.6 cm^2 ($A/\sqrt{K}$) using temperature sensors. Detecting leakage starting from 0.3 cm^2 ($A/\sqrt{K}$) assures the setpoint calculation will have margin to the 0.6 cm^2 ($A/\sqrt{K}$) analytical limit to close a VBIV." GEH stated that "[t]he temperature difference value that will activate the sensors will be dependent on the final location of the temperature sensors, the instrument accuracy of the temperature sensors, and the height of the vacuum breaker seat from the diaphragm floor, which is dependent on the end-to-end dimension of the VBIV."

In RAIs 6.2-148 S02 and S03, staff asked GEH to provide details of the type test and how the setpoint will be determined. In response GEH submitted licensing topical report, NEDE-33564P, "Leakage Detection Instrumentation Confirmatory Test for the ESBWR Wetwell-Drywell Vacuum Breakers," providing details of the type test and the method of determining the setpoint and agreed to incorporate this report by reference in DCD Tier 2, Revision 8. After reviewing the GEH responses including NEDE-33564P, staff finds that GEH responses address staff's concerns and are acceptable.

Loss-of-Coolant Accidents. The staff reviewed the information provided in DCD Tier 2, Section 6.2.1.1 and performed an audit of the GEH containment analysis on December 11 through December 15, 2006. In addition, the staff performed confirmatory containment analyses using the MELCOR computer code that produced qualitative agreement with those of GEH.

Treatment of Noncondensable Gases

The stratification and holdup of noncondensable gases in the drywell during the blowdown phase of the LOCA and their later release can affect the performance of the PCCS. If the performance of the PCCS during the long-term cooling phase of the LOCA is degraded because of the presence of noncondensable gases that were not purged during the blowdown, then the steam that is not condensed in the PCCS will be vented to the suppression pool. This raises the temperature of the suppression pool and increases the containment pressure.

The NRC-approved approach addresses uncertainties in the ability of TRACG to account for mixing and stratification in the drywell (NEDC-33083P-A). The NRC-approved TRACG model consisted of a "tee" model to control the release of noncondensable gases from the lower drywell (NEDC-33083P-A and NEDE-32176P). The DCD model does not have such a "tee" model to control noncondensable gases, and the DCD does not describe the behavior of noncondensable gases. It appears that a newer model was used for the containment analysis presented in the DCD. Therefore, in RAI 6.2-52, the staff requested that GEH provide justification for the modeling changes and a discussion of containment response to the limiting DBA with respect to noncondensable gas holdup, movement, mixing, and stratification throughout the containment. The staff needed this information to determine whether noncondensable gas mixing and stratification in the containment are appropriately modeled in the evaluation of the ESBWR containment performance. In response, GEH described the modeling changes and the results of tieback calculations performed to determine the effect of the modeling changes; the impact on containment performance from the modeling changes was minimal. GEH described the behavior of noncondensable gases in the containment adequately. However, GEH did not provide justification for modeling changes. RAI 6.2-52 was being tracked as an open item in the SER with open items.

In a supplemental request to RAI 6.2-52, the staff asked GEH to justify modeling changes and provide the justification and the results of the tieback calculations in the DCD or in a supplement to NEDC-33083P-A. In response, GEH added Appendix B to DCD Tier 2, Revision 5, justifying modeling changes and providing results of the tie-back calculations. GEH stated that the analysis for the ESBWR containment evaluation followed the application methodology outlines in NEDC-33083P-A and that TRACG nodalization approach in the licensing analysis was similar to that used in NEDC-33083P-A. GEH stated that this licensing nodalization includes additional features and details. Some of these features were to address the confirmatory items listed in the safety evaluation report of NEDC-33083P-A and others were implemented due to design changes. GEH added Table 6.2-6a to DCD Tier 2, Revision 4 summarizing the list of these changes in the TRACG nodalization. GEH addressed ESBWR design changes which were made after staff evaluated NEDC-33083P-A as described in the corresponding SER. Therefore, the staff finds that RAI 6.2-52 is resolved.

DCD Tier 2, Revision 1, did not discuss the containment response to the limiting DBA with respect to the movement of noncondensable gases and mixing and stratification in the containment. This information is needed for the review of the containment performance in response to the limiting DBA. Therefore, in RAI 6.2-53, the staff requested this information. In response, GEH provided the results of nominal analysis for the limiting DBA. The staff makes its determination on containment performance based on bounding analysis but not on nominal analysis. Therefore, in RAI 6.2-98, the staff asked GEH to update its response to RAI 6.2-53 by performing bounding analysis.

Also, because the limiting DBA changed from the FWLB to the MSLB as discussed in RAI 6.2-59 (above), the staff requested in a supplement to RAI 6.2-53 that GEH reanalyze the containment response to MSLB as the limiting DBA. In response, GEH added the results of containment response to the limiting DBA, with respect to the movement of noncondensable gases and mixing and stratification in the containment for the FWLB and MSLB scenarios.

RAI 6.2-53 and RAI 6.2-98 were being tracked as open items in the SER with open items. The applicant's response addresses the staff's concern and is acceptable because the applicant's treatment of noncondensable gases is bounding. Based on the applicant's responses, RAI 6.2-59 and RAI 6.2-98 are resolved.

Treatment of Nonsafety-Related Systems

DCD Tier 2, Section 19A.3.1.2, describes the ESBWR treatment of nonsafety systems. The safety-related ICS and the safety-related PCCS provide the safety function of removing reactor decay heat from the core and containment. These systems are capable of removing decay heat for at least 72 hours without the need for active systems or operator actions. After 72 hours, makeup water is needed to replenish the boil-off from the upper containment pools. The ESBWR design includes permanently installed piping in the FAPCS that connects directly to a diesel-driven makeup pump system. This connection enables the upper containment pools and spent fuel pools to be filled with water from the fire protection system (FPS), which provides onsite makeup water to extend the cooling period from 72 hours to 7 days. The dedicated FPS equipment for providing makeup water and the flow paths to the pools are classified as nonsafety-related. A dedicated external connection to the FAPCS line allows for manual hookup of external water sources, if needed, at 7 days for either upper containment pool replenishment and for spent fuel pool makeup. These functions are manually actuated from the yard area and can be performed without any support systems. The components within the scope of regulatory treatment of nonsafety systems (RTNSS) are the diesel-driven makeup pump system, FAPCS piping connecting to the diesel-driven makeup pump system, and the external connection.

DCD Tier 2, Revision 1, was not clear as to whether the containment analysis takes credit for the nonsafety systems. Therefore, in RAI 6.2-57, the staff asked GEH to discuss the effect of the nonsafety systems in the mass and energy released into the containment and how these systems would respond during the DBAs analyzed (FWLB, MSLB, GDCS line break, and bottom drain line break). In response, GEH stated that the ESBWR took no credit for the nonsafety systems for the ECCS and containment analyses. GEH summarized the nonsafety systems and described their functions and impact on the LOCA responses, if they are available. These systems are the high-pressure CRD system, reactor water cleanup/shutdown cooling system (RWCU/SDC), FAPCS in suppression pool cooling mode, FAPCS in drywell spray mode, and FAPCS in low-pressure coolant injection (LPCI) mode. GEH updated the DCD to include this information. The staff confirmed that the information was incorporated in DCD Tier 2, Revision 5.

RAI 6.2-57 was being tracked as an open item in the SER with open items. The applicant's response addresses the staff's concern and is acceptable because the applicant's explanation of the treatment of nonsafety systems is satisfactory. Based on the applicant's response, RAI 6.2-57 is resolved.

Maximum Containment Pressure

The staff noticed that the containment pressure predicted for the limiting DBA continued to increase until the end of the calculation time of 72 hours following a LOCA, with a possibility of exceeding the containment design pressure after 72 hours. The section below titled "Post-72-Hour Containment Pressure Control," discusses this issue.

DCD Tier 2, Revision 3, Section 6.2.1.1.3.5, states that "the peak drywell pressure for the bounding case is below the containment design pressure." DCD Tier 2, Revision 3, Table 6.2-5, lists peak drywell pressure and peak wetwell pressure. However, the TRACG analysis results provided in the DCD show no peak drywell or wetwell pressures for the limiting FWLB and MSLB DBAs. Instead, the pressure continues to rise and reaches its maximum value for the duration of analysis at 72 hours as stated above. In RAI 6.2-177, the staff requested that GEH correct this discrepancy. In response to this RAI, GEH changed references to "peak pressure" to "maximum pressure" in DCD Tier 2, Revision 4. The staff confirmed that the change was incorporated in DCD Tier 2, Revision 4.

RAI 6.2-177 was being tracked as an open item in the SER with open items. The applicant's response addresses the staff's concern and is acceptable because the applicant revised DCD Tier 2 as requested. Based on the applicant's response, RAI 6.2-177 is resolved.

Single Failures Considered

DCD Tier 2, Revision 1, did not describe the active single failures considered when analyzing the containment performance under DBAs. As stated in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Revision 3, Section 6.2.1.4, a failure mode and effects analysis should be performed to determine the most severe single active failure for each break location for the purpose of maximizing the mass and energy released to the containment and the containment pressure response. The analysis should consider, for example, the failure of a steam or feedwater isolation valve, the feedwater pump trip, and containment heat removal equipment. Therefore, in RAI 6.2-58, the staff asked GEH to discuss the active single failures considered for each break type (FWLB, MSLB, GDCS line break, and vessel bottom line break) and to provide the resulting peak pressure and temperature for each case evaluated using appropriate licensing analysis assumptions to conservatively maximize the containment pressure or temperature response for each case.

In response, GEH stated that DCD Tier 2, Table 6.3-6, summarizes the single, active failures considered in the ECCS performance analysis. The assumed single failures are one DPV, one SRV, and one GDCS injection valve. Other postulated failures are not specifically considered, because they all result in at least as much ECCS capacity as one of the above failures. The assumed single failures for the containment analysis are one DPV and one SRV. Results of double-ended guillotine (DEG) pipe break analyses at four different locations show that an instantaneous guillotine rupture of an MSL with failure of one DPV produces the most limiting responses for the containment pressure evaluation. The second limiting case is an instantaneous guillotine rupture of an FWL with failure of one SRV.

The GEH response states that various single active failures were considered in the ECCS analysis. However, it was not clear whether the single failures considered would bound the single failures affecting the maximum containment pressure. For example, an MSLB or FWLB with a failure of a shutoff valve in one of the standby liquid control system (SLCS) trains was not considered for peak containment pressure and temperature analysis. DCD Tier 2, Section 9.3.5.2, states that the operation of the accumulator vent could limit the amount of nitrogen injected into the reactor vessel by assisting in reducing accumulator pressure. However, if a shutoff valve in one of the SLCS trains fails, nitrogen could be transported to the reactor vessel until the accumulator tank depressurizes (with the assistance of the accumulator vent). The effect of this event on the peak ESBWR containment pressure was not analyzed. Therefore, in a supplement to RAI 6.2-58, the staff requested that GEH describe the active single failures considered with respect to peak containment pressure.

In response, GEH stated that to avoid the injection of nitrogen into the reactor vessel, four divisional, safety-related level sensors per SLC accumulator are used to provide automatic isolation of the associated accumulator shutoff valves (two in series) on a low accumulator level signal, using a two-out-of-four voting logic as stated in DCD Tier 2, Section 7.4.1.2. Therefore, the staff finds that a failure of a shutoff valve in one of the SLCS trains will not cause continuous injection of nitrogen in the pressure vessel and need not be considered as a credible single failure for containment analysis.

RAI 6.2-58 was being tracked as an open item in the SER with open items. The applicant’s response is acceptable because the staff finds that the single active failures considered by GEH produced the highest maximum containment pressure. Based on the applicant’s response, RAI 6.2-58 is resolved.

Initial Containment Conditions

DCD Tier 2, Table 6.2-2, lists the average drywell temperature during normal operation as 57.2 degrees C (135 degrees F). However, DCD Tier 2, Table 6.2-6, lists the initial temperature used in analyzing the containment DBA cases as 46.1 degrees C (115 degrees F). In RAI 6.2-64, the staff asked GEH to justify its position that the lower-than-average drywell temperature during normal operation used in the containment analysis would provide conservative results. GEH responded that the expected operating range of drywell temperature is from 46.1 degrees C (115 degrees F) to 57.2 degrees C (135 degrees F). GEH also discussed results from a previous sensitivity study of the simplified boiling-water reactor (SBWR) design that showed that increasing initial drywell temperature caused a decrease in the long-term drywell pressure. Cooler initial temperature represents more initial inventory for the noncondensable gases and, consequently, higher long-term containment pressure. Therefore, the reported DBA analyses were performed at 46.1 degrees C (115 degrees F) to ensure conservative (i.e., maximum) calculated peak drywell pressure. GEH agreed to update the DCD to include this response.

RAI 6.2-64 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the change was incorporated in DCD Tier 2, Revision 5. Based on the applicant’s response, RAI 6.2-64 is resolved.

DCD Tier 2, Table 6.2-2, lists the average drywell relative humidity during normal operation as 50 percent. However, DCD Tier 2, Table 6.2-6, lists the initial relative humidity used in analyzing the containment DBA cases as 20 percent. In RAI 6.2-65, the staff asked GEH to justify its statement that the lower-than-average drywell relative humidity during normal operation used in the containment analysis would provide conservative results. GEH responded that the lower bound on the relative humidity in the drywell is 20 percent. It selected the lower bound value because a lower initial drywell relative humidity results in more noncondensable gases available to be transferred to the wetwell and higher containment pressures following the LOCA. GEH agreed to update the DCD to include this response.

RAI 6.2-65 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the change was incorporated in DCD Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-65 is resolved.

DCD Tier 2, Table 6.2-2, lists the suppression pool temperature in hot standby as 54.4 degrees C (130 degrees F), while DCD Tier 2, Table 6.2-6, lists the initial suppression pool temperature used for the DBA analyses as 43.3 degrees C (110 degrees F), which is lower than the hot standby temperature. In RAI 6.2-67, the staff asked GEH (1) to justify that the suppression pool initial temperature used for the containment analysis would provide conservative results and (2) to describe the impact of operating the reactor at less than 100-percent power with respect to the stored energy and mass in the primary system which would be released to containment during a DBA.

Regarding initial pool temperature, GEH stated that the suppression pool average temperature during normal operation was less than 43.3 degrees C (110 degrees F), and the maximum pool temperature of 43.3 degrees C (110 degrees F) was used in the safety analyses. According to the TS (DCD Tier 2, Chapter 16), the reactor is required to reduce thermal power to less than 1 percent of rated thermal power when the suppression pool temperature is greater than or equal to 43.3 degrees C (110 degrees F), and the reactor will be switched to shutdown mode immediately when the suppression pool temperature is greater than or equal to 48.9 degrees C (120 degrees F).

Regarding the second concern, the mass and energy releases in the case of a reactor operating at less than 100-percent power are bounded by those for 100-percent power scenarios, and, therefore, are less severe than the limiting DBA case.

RAI 6.2-67 was being tracked as an open item. The applicant's response is acceptable because the applicant's choice of initial suppression pool temperature is consistent with relevant TS. Based on the applicant's response, RAI 6.2-67 is resolved.

TRACG Modeling Parameters

In the "Pre-application Model," as described in Section 3.3.1.1.1 of NEDC-33083P, GEH conservatively modeled the suppression pool by forcing energy entering the pool to mix with and heat only the portion of the pool above the level of entry. This was accomplished by restricting the flow area of the suppression pool cells below the source of energy addition. The DCD was not clear as to whether the same model was used for the analysis presented in the DCD. Therefore, in RAI 6.2-55, the staff requested clarification from GEH. In response, GEH stated that it had used the same approach for all the DCD calculations, except for FWLB. Following an FWLB, energy addition from the spillover continues in the long-term heatup, so the flow area restriction is not applied. The staff finds that because of the long-term energy addition to the pool by spillover flow following an FWLB, the exception for FWLB is acceptable. However, the applicant modified the design by removing the spillover pipes and accomplishing the spillover function by spillover horizontal holes, which is reflected in DCD Tier 2, Revision 3, Section 6.2.1.1.2, thus invalidating the above concern. Therefore, RAI 6.2-55 is resolved.

In RAI 6.2-63, the staff asked GEH to provide (1) the energy source information identified in RG 1.70, Table 6.9, for the limiting FWLB and limiting MSLB cases and (2) energy removal by the PCCS. This information is needed for proper review of the TRACG analyses, as well as for the staff's performance of confirmatory containment analysis using the MELCOR computer code. GEH provided the requested information in the DCD Tier 2, Revision 5, Section 6.2.1.3, and added DCD Tier 2, Table 6.2-12d and Figures 6.2-9e1, 6.2-9e2, 6.2-10e1, and 6.2-10e2.

RAI 6.2-63 was being tracked as an open item in the SER with open items. The applicant's response is acceptable because the applicant revised DCD Tier 2 as requested. Based on the applicant's response, RAI 6.2-63 is resolved.

Previous versions of the DCD did not contain information on how GEH evaluated the various containment volumes to ensure a conservative evaluation of the containment response to DBAs. These volumes include gas space in the drywell, wetwell, and GDCS pool and water volume in the suppression and GDCS pools. Therefore, in RAI 6.2-69, the staff asked GEH to provide this information.

In response, GEH stated that it had calculated the net drywell gas space volume by subtracting the displaced volumes occupied by equipment and structures located inside the drywell from the gross drywell volume. The gross drywell volume is calculated from the available arrangement drawings. GEH calculated the displaced volumes of equipment and structures, including the RPV, reactor shield wall (RSW), GDCS pool structures, RPV support brackets, fine motion CRDs, and the protective layer on basemat, from the design drawings. GEH assumed, based on engineering judgment, that the other piping, equipment, and miscellaneous structures would displace a total of 1 percent of the gross volume.

GEH calculated the net wetwell gas space volume by subtracting the displaced volume occupied by the equipment hatches that are located in this region from the gross volume. GEH assumed that the displaced volume occupied by the equipment hatch is 0.1 percent of the total gross volume. GEH calculated the net gas space volume above the GDCS pools from the gross volume, assuming insignificant volume compared to the total gross volumes for other equipment and structures located in these regions. GEH calculated the gross wetwell volume from the available arrangement drawings. GEH calculated the net GDCS pool water volumes (total volume and nondrainable volume) from the available arrangement drawings and GDCS drain pipe suction elevation.

GEH calculated the net suppression pool water volume from the available arrangement drawings and assumed insignificant volume as compared to the total gross volumes for other equipment and structures located in these regions. In response to RAI 6.2-69, GEH revised DCD Tier 2, Table 6.2-6, and DCD Tier 1, Table 2.15.1-2. The revision specified the maximum and minimum analytical values for drywell and wetwell volumes used in the licensing analyses, and the inspection, test, analysis, and acceptance criteria (ITAAC) ensure that the as-built volumes match or are conservative with respect to the containment performance analysis.

RAI 6.2-69 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the change was incorporated in DCD Tier 1 and Tier 2, Revision 6. Based on the applicant's response, RAI 6.2-69 is resolved.

Previous versions of the DCD did not include information on how GEH evaluated the various primary system volumes and heat structures (piping, RPV, and others). DCD Tier 2, Table 6.2-6, provided the reactor power and reactor pressure for the bounding case but not the reactor temperature. The staff needs this information to determine whether these values were conservatively evaluated. In RAI 6.2-70, the staff requested that GEH provide this information. In response, GEH described how it evaluated primary system volumes and heat structures using the available design drawings. Regarding the reactor temperature used for the containment analysis, GEH stated that the reactor dome temperature corresponds to the saturation temperature at the specified dome pressure. RAI 6.2-70 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD Tier 2, Section 6A and Table 6.2-6. Based on the applicant's response, RAI 6.2-70 is resolved.

GDCS Airspace

DCD Tier 2, Section 6.2.1.1.10.2, states that the GDCS pools are placed above the RPV with their airspace connected to the drywell, and that once the GDCS pools are drained, the total volume of the GDCS pools is added to the volume of the drywell airspace. The staff believes that adding volume to the drywell airspace was not possible because the water removed from the GDCS pools would occupy the drywell volume. In RAI 6.2-152, the staff requested an explanation from GEH. RAI 6.2-152 was being tracked as a confirmatory item in the SER with open items. In response, GEH concurred that the statement was misleading because there was no net gain of drywell airspace resulting from the draining of the GDCS pools. GEH deleted the statement from the DCD in a later revision, and the staff confirmed the change. Based on the applicant's response, RAI 6.2-152 is resolved.

TRACG Modeling

The TRACG model used for the analysis presented in the DCD has an additional axial node in the upper wetwell that is not in the model used in preapplication, which was reviewed by the staff. In the preapplication TRACG model, the treatment of the upper wetwell limited mixing to conservatively assess the wetwell gas space temperature. In RAI 6.2-54, the staff asked GEH to (1) provide the rationale for adding the additional axial node, (2) state whether the same conservative approach used in the preapplication TRACG model was used in the DCD TRACG model, and (3) state whether the gas space temperature was treated conservatively. In response, GEH stated that there are 24 I-beams located at the top of the wetwell to support the diaphragm floor, and an additional axial node was added to the wetwell to refine the simulation of the trapped gas space between the I-beams. GEH stated that it had used the same conservative approach described in the preapplication model in the DCD TRACG model. GEH stated that the gas space temperature was treated in a conservative manner as described in the preapplication report. It applied an irreversible loss coefficient at the interface between the cells in the top two gas space levels to introduce forced stratification, thereby restricting flow between cells in the top two gas space levels. GEH agreed to add this information in a later revision of DCD Tier 2.

RAI 6.2-54 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the information was incorporated in DCD Tier 2, Revision 5, Appendix 6B. Based on the applicant's response, RAI 6.2-54 is resolved.

The original DCD did not provide information on passive heat sinks used in the containment analysis. The staff needed this information to perform confirmatory containment analysis. Therefore, in RAI 6.2-62, the staff asked GEH to provide this information as listed in RG 1.70, Table 6-11, per SRP Section 6.2.1.1.C. RAI 6.2-62 was being tracked as an open item in the SER with open items. The applicant provided the requested information in Appendix 6D to DCD Tier 2, Revision 3. Based on the applicant's response, RAI 6.2-62 is resolved.

The applicant identified the systems modeled as part of the DCD version of the TRACG model but did not show them in the nodal scheme. The staff needed a more complete nodalization, including, for example, the ICS, the SLCS, and the feedwater system, to review the TRACG model. Therefore, in RAI 6.2-72, the staff requested that GEH provide this information. In response, GEH provided the TRACG nodalization schematic diagrams for the ICS and feedwater system, which were later added to the DCD. GEH stated that the SLCS was simulated via a FILL component (FILL0037) that injected boric liquid into the RPV at the mid-elevation of the outer bypass (RPV axial Level #5, Ring #3). GEH agreed to update the DCD to include this information.

RAI 6.2-72 was being tracked as a confirmatory item. The staff confirmed that GEH added the modeling information for the SLCS in Appendix B to DCD Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-72 is resolved.

The DCD was not clear as to (1) how GEH applied the $\pm 2\sigma$ uncertainty to the choked flow in lines, SRVs, DPVs, and both sides of breaks and (2) which critical flow models were used for choked flow paths. Because the staff needed this information for its review, in RAI 6.2-73, the staff asked GEH to provide this information. In response to part (1) of the request, GEH stated that the upper limit ($+2\sigma$) is applied to the bounding short-term peak pressure calculations, and the lower bound (-2σ) is applied in the long-term peak pressure calculations. This response is acceptable because the chosen uncertainty values for the choked flow provide conservative results for accident scenarios, which have bounding short-term or long-term peak pressure. However, as stated below under resolution of RAI 6.2-59, after error corrections in TRACG calculations, no accident scenario showed bounding short-term peak pressures. In RAI 6.2-141 the staff requested that GEH revise all previous responses to the containment-related RAIs, which includes RAI 6.2-73.

In response to part (2) of RAI 6.2-73, GEH stated that the TRACG critical flow model was applied to all flow paths at locations where the choking calculation was specified in the input model. These choked paths included the SRVs, DPVs, FWLB (RPV side), FWLB (balance-of-plant side), and drywell main vents. The staff finds that applying the choked flow model to all flow paths was reasonable and acceptable. In response, GEH agreed to update the DCD to provide information submitted in response to RAI 6.2-73. RAI 6.2-73 and RAI 6.2-141 were being tracked as open items in the DCD with open items.

In response to RAI 6.2-141, the applicant stated that all of the most recent containment analyses confirmed the MSLB scenario as the bounding case, as documented in DCD Tier 2, Revision 3, Section 6.2. The staff finds that this response is acceptable because it addressed the staff's concern. RAI 6.2-141 is resolved. GEH updated DCD Tier 2, Revision 5, Section 6.2 to provide information submitted in response to RAI 6.2-73. RAI 6.2-73 is resolved.

TRACG Results

Previous versions of the DCD provided the results in graphic form only for FWLB, but not for GDCS line break, vessel bottom line break, or MSLB. The staff needed the results for these other breaks for its review of containment response to DBAs. Therefore, in RAI 6.2-59, the staff asked GEH to provide this information. In response, GEH provided graphical results of FWLB, GDCS line break, vessel bottom line break, and MSLB. Each of these cases considered a single failure and nominal conditions given on Table 6.2-6 of DCD Tier 2, Revision 1, and assumed 100 percent double-ended guillotine break. In its response GEH agreed to include above results in the DCD. After reviewing the results, the staff finds that they are acceptable.

However, in its response to RAI 6.2-59, GEH also stated that it had discovered an erroneous result for FWLB (i.e., an early peak in drywell pressure), because the FWLB analysis was sensitive to the time step selection. GEH found that the pressure disturbance was the result of a numerical problem, commonly known as "water packing." Water packing generally occurs when steam is condensing in the subcooled water in a confined volume. Usually, this numerical problem can be avoided by using smaller time steps during the period when the water packing problem is likely to occur. Lowering the time step from 0.05 to 0.025 corrected this problem. GEH also stated that it had corrected three input errors in vacuum breaker flow area, SLCS flow input table, and axial power input into part-length fuel rods and enhanced models for vapor additive friction loss coefficients. GEH revised the analysis presented in NEDC-33083P-A, reflecting the correction of the error and model enhancement applied to FWLB, GDCS line break, vessel bottom line break, and MSLB, and updated the DCD. The staff finds that the GEH error corrections as described in its response are acceptable. In RAI 6.2-59 S01 the staff requested GEH include the input error corrections information in a licensing document. RAI 6.2-59 S01 was being tracked as an open item in the SER with open items.

In response, GEH added the input error corrections information to Appendix B to ESBWR DCD Tier 2, Revision 4. This addressed the staff's concern. RAI 6.2-59 S01 is resolved.

DCD Tier 2, states that only DEG breaks were analyzed. However, the DCD also states that a spectrum of break sizes was evaluated but does not describe the results. The information on containment analysis for breaks smaller than DEG breaks is needed to confirm that the four DEG breaks analyzed (FWLB, GDCS line break, vessel bottom line break, and MSLB) were limiting DBAs. Therefore, in RAI 6.2-60, the staff requested that GEH (1) confirm whether only four DEG breaks with different locations and sizes were analyzed, (2) provide the results of sensitivity analyses for smaller than DEG break sizes for FWLB and MSLB to ensure that DEG breaks were limiting, and (3) provide the results of sensitivity analyses for MSLB at high and low locations in the containment to justify that the MSLB analyzed was limiting.

In response to part (1) of the request, GEH clarified that it had performed containment design-basis calculations for a spectrum of four DEG pipe break sizes and locations to ensure that it had identified the worst case and updated the DCD to include this clarification. In response to part (2) of the request, GEH provided and described the results of parametric analyses performed with different break areas (40 percent, 60 percent, 80 percent, and 100 percent of the DEG break area) for FWLB and MSLB. These analyses showed that the breaks with 100 percent of the DEG break areas were limiting. This confirmed that the assumed 100-percent DEG break size for the DBA MSLB analysis was limiting. In response to part (3) of the request, GEH provided and described results of the base-case calculation performed for a break occurring in the drywell at Level 34 as shown in DCD Tier 2, Figure 6.2-7, and parametric calculations for breaks occurring at Levels 31, 25, and 23. The base case with the highest break location generated the highest maximum drywell pressure. This confirmed that the base-case break location assumed for the DBA MSLB analysis was limiting. After reviewing the GEH response, the staff finds that it is acceptable because it addressed the staff's concerns. In RAI 6.2-60 S01 the staff requested GEH to incorporate the response into the DCD. RAI 6.2-60 S01 was being tracked as an open item in the SER with open items.

In response GEH added a discussion of spectrum of break sizes and break elevations as DCD Tier 2, Appendix 6F. This addressed the staff's concerns. RAI 6.2-60 S01 is resolved.

For the DBAs analyzed, ESBWR DCD Tier 2 did not provide mass and energy release data, mass inventories for systems modeled, and gas and pool stratification data. The staff needs this information for its review of TRACG containment analysis. Therefore, in RAI 6.2-61, part 1, the staff asked GEH to provide mass and energy release data from the RPV side and from the balance-of-plant side of the break for the limiting FWLB and limiting MSLB.

In RAI 6.2-61, part 2, the staff requested that GEH provide, for the limiting FWLB and limiting MSLB, (a) mass and energy release from the safety valves and DPVs, (b) mass flow through GDCS, PCCS, ICS, SLCS, hydraulic control units (HCUs), drywell main vents, wetwell to drywell vacuum breakers, and drywell leakage, (c) RPV water level-collapsed and two-phase, drywell pool level, suppression pool level, GDCS water level, PCCS/ICS upper pool level, noncondensable partial pressure in the drywell and wetwell, (d) local gas and pool temperatures in the drywell, wetwell, and RPV to reveal regional stratification for selected nodes, and (e) suspended liquid water masses for the RPV steam dome, drywell, and wetwell volumes.

During an NRC audit conducted December 11–15, 2006, GEH stated that it had made several changes to the TRACG containment model. GEH identified these changes in DCD Tier 2, Revision 3, Appendix 6A. GEH made a design configuration change to designate feedwater isolation as safety grade, which made MSLB the limiting DBA for containment performance. GEH supplemented its response to RAI 6.2-61 by providing nominal and bounding analyses for the MSLB. The staff used the information provided in response to RAI 6.2-61 to perform confirmatory containment performance analysis with the MELCOR computer code. The response is acceptable because the applicant provided the revised results for FWLB and MSLB cases and modified DCD Tier 2 accordingly. Based on the applicant's response, RAI 6.2-61 is resolved.

Earlier versions of the ESBWR DCD provided predictions for containment temperature in graphs of temperature versus time for DBAs analyzed. However, GEH did not provide information on how the temperatures were combined to determine the values shown in the graphs, because the DCD version of the TRACG model was nodalized for the free volumes and pool regions. The staff needs this information to compare its confirmatory containment analysis results with the GEH results. Therefore, in RAI 6.2-71, the staff requested that GEH provide this information. In response, GEH stated that the temperatures provided represent the maximum envelope of the corresponding temperatures from all the cells residing in the region of interest.

GEH stated that individual cell temperatures would better describe the response to thermal stratification (such as that in the suppression pool and in the wetwell). GEH updated the graphs in the DCD to identify cells for which temperatures plotted. The applicant provided the requested information, in support of the staff's confirmatory calculations, and revised the DCD accordingly. Based on the applicant's response, RAI 6.2-71 is resolved.

In figure titles, DCD Tier 2 incorrectly referred to noncondensable gas as “air.” For example, see DCD Tier 2, Revision 6, Figure 6.2-14d1, “Main Steam Line Break (Bounding Case)—Drywell and GDCS Air Pressures (72 hrs).” In RAI 6.2-176, the staff requested that GEH correct this. GEH made the requested editorial changes replacing all “GDCS Air Pressures” captions with “GDCS NC Gas Pressures.” RAI 6.2-176 was being tracked as an open item in the SER with open items. The staff confirmed that these changes were incorporated in DCD Tier 2, Revision 7. Based on the applicant’s response, RAI 6.2-176 is resolved.

Post-72-Hour Containment Pressure Control

ESBWR DCD Tier 2 provides TRACG results for up to 72 hours following the initiation of a LOCA. The maximum drywell pressure predicted by TRACG for the limiting DBA of MSLB is 384.2 kPa absolute (55.8 pounds per square inch absolute [psia]), which is 29.0 kPa (4.2 psi) below the containment design pressure of 411.7 kPa absolute (59.7 psia) (i.e., 310 kPaG [45 psig]). However, the maximum drywell pressure is predicted to occur at 72 hours, when the calculation ends, and the pressure increases continually with a possibility of exceeding the design pressure post-72 hours. GDC 50 requires the containment and its associated systems to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

The staff’s concern about the long-term cooling capability was the subject of RAI 6.2-140 and RAIs 6.2-140 S01-S06. RAI 6.2-140 and RAIs 6.2-140 S01-S06 were being tracked as open items in the SER.

GEH, in a series of responses, proposed assuming that the following occur beginning 72 hours after initiation of a LOCA to reduce the long-term containment pressure:

- Continuous refilling of the PCCS pools at a rate commensurate with decay heat rate,

- Taking credit for the passive autocatalytic recombiners and removing from the system hydrogen at the rate of its generation, and

- Implementing a design modification by installing vent fans, teed off of each PCCS vent line, thus establishing a sufficient gas flow from the DW atmosphere to the exhausts submerged in the GDCS pool. This fan system is to be designed to satisfy minimum requirements such as to assure the long term removal of noncondensable gas from the PCCS for continued condenser efficiency.

With these modifications, the calculated containment pressure drops rapidly shortly after 72 hours of the postulated limiting DBA from the maximum pressure to about 330 kPa absolute (47.8 psia), and continues to decrease over the period of 30 days to about 290 kPa absolute (42.1 psia). Thus, during the whole 30-day period following a LOCA the predicted containment pressure remained below the containment design pressure of 411.7 kPa absolute (59.7 psia). After reviewing the proposed design modifications the staff finds them acceptable.

The results of staff’s confirmatory calculations using MELCOR computer code showed similar results as the GEH TRACG calculation. These addressed the staff’s concerns. RAI 6.2-140 is resolved.

Staff Audit of TRACG Containment Analysis

The staff audited the GEH TRACG containment analysis on December 11–15, 2006. The following is a summary of the staff's observations and their resolution.

The amount of noncondensables in the GDCS airspace is sensitive to whether a single pipe node or a double pipe node is used in modeling the junction between the GDCS airspace and the drywell. GEH later changed the TRACG nodalization to use a double pipe junction for bounding DBA containment analyses.

The staff requested that GEH update the TRACG LOCA application to the ESBWR by considering the modeling changes that have been made since the original approval. GEH agreed and later provided this information as Appendix A to DCD Tier 2, Chapter 6.

DCD Tier 2, Revision 2, Section 6.3, assumed the availability of the containment back pressure in determining the minimum water level in the RPV following a LOCA. The depressurization of the RPV and thus the initiation of the GDCS depends on the assumptions used for determining the containment back pressure. However, the GEH analyses were inconsistent with SRP Section 6.2.1.5, Revision 3, and the associated Branch Technical Position (BTP) CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Although CSB 6-1 was developed to evaluate the performance of the ECCS of a pressurized-water reactor (PWR), most of its guidance also applies to determining the performance of the GDCS for the ESBWR. Specifically, the input information for the model, active heat sinks (e.g., FAPCS operating in drywell spray mode), and passive heat sinks affect the containment back pressure. During the audit, the staff asked GEH to justify the containment back pressure used for determining the minimum RPV water level considering BTP CSB 6-1. The staff requested this information in RAI 6.2-144. In response, GEH evaluated the impact of containment back pressure on the ECCS performance and presented this evaluation in ESBWR DCD Tier 2, Revision 4, Appendix 6C. The staff reviewed the applicant's evaluation and finds that the minimum chimney collapsed level is not sensitive to the changes in the containment back pressure expected for the ESBWR design under LOCA conditions. Based on the applicant's response, RAI 6.2-144 is resolved.

Staff Confirmatory Analysis

The staff used the MELCOR computer code to perform confirmatory analysis for the ESBWR DBA containment performance evaluation for the bounding MSLB scenario as presented in DCD Revision 3. The MELCOR model was set up using the bounding initial and model parameters and biases as described in the DCD and GEH responses to staff's RAIs. The MELCOR model used a well-mixed drywell volume, resulting in minimal noncondensable gas trapping.

Table 6.2-2 in this report lists a sequence of events and compares the predicted timing of events. Automatic depressurization system (ADS) actuation agreement is within a few seconds between the DCD reported time and those times calculated with the MELCOR model. MELCOR predicted that the expansion/passive containment cooling (PCC) tank reflood would occur 34,376 seconds (9.55 hours) earlier than predicted by TRACG. However, the reflood timing has a small impact on containment pressure responses since the PCCS efficiency is not notably affected by the relatively small amount of tube length uncovered before reflood (about one-fourth uncovered).

The difference in the reflood timing is the result of differences between the TRACG and MELCOR models relative to the trapping of drywell gases and, subsequently, the rate of release of those gases to the PCCS. The TRACG and MELCOR event timings agree reasonably well.

Table 6.2-2. Sequence of Events for MSLB (Bounding Case) with Failure of One DPV.

Event	Time (s)	
	DCD Tier 2, Revision 3	MELCOR
Guillotine break of MSL inside containment	0	0
Main vents clear Top vent: Middle vent: Bottom vent:	 1.8 2.3 3.1	 1.1 1.6 2.8
Reactor isolated	13	13
Level 1 is reached	496	482
Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated	506	492
DPV actuation begins at 50 s after confirmed Level 1 signal; SLCS flow starts	556	542
GDCS flow into vessel begins	726	686
SLCS flow is depleted	856	832
PCC pool drops below the elevation of 29.6 m (97.1 ft) ; water from dryer/storage pool flows into expansion pool	126,776	92,400
Drywell pressure attains peak	259,000 ~72 h (384.2 kPa [55.71 psia])	259,000 ~72 h (370.5 kPa [53.72 psia)

Table 6.2-3 summarizes the maximum pressures calculated and their margins to design pressure for the bounding MSLB scenario using TRACG and MELCOR computer codes. Margin to design pressure is defined as $(P_d - P)/P_d$, where P_d is the design gauge pressure and P is the calculated gauge pressure. Both TRACG and MELCOR predicted the maximum pressure occurring at 72 hours following an MSLB. The comparisons of pressure profiles between the DCD and MELCOR calculation for the bounding MSLB case are quite good if the blowdown period can be excluded.

However, as the licensing focus moves from blowdown to later times, such as the GDCS recovery period and long-term cooling, the pressures reported in the DCD and calculated with MELCOR are essentially equivalent. At 72 hours, the DCD-reported drywell pressure of 384.2 kPa absolute (55.7 psia) and the MELCOR-calculated pressure of 370.5 kPa absolute

(53.7 psia) provide reasonable confirmation of the certification analysis presented in the DCD. Margins to design pressure for the DCD and MELCOR calculation are 8.9 and 13 percent, respectively.

Table 6.2-3. Summary of Peak Pressures Calculated for the Bounding MSLB Scenario Using TRACG and MELCOR Computer Codes.

Case	TRACG (DCD Rev. 3)		MELCOR	
	Pressure (kPa absolute)	Margin to Design Pressure (%)	Pressure (kPa absolute)	Margin to Design Pressure (%)
Reference	384.2 (55.7 psia)	8.9	370.5 (53.7 psia)	13
Radiolytic gas generation terminated at 12 h	---	---	347 (50.3 psia)	21
Bypass leakage doubled	---	---	400 (58.0 psia)	3.8

Table 6.2-3 also presents the results of the MELCOR calculations performed to address the long-term pressurization sensitivity to radiolytic gas source and bypass leakage area. The doubling of the bypass leakage capacity reduced the margin to the design pressure from 13 to 3.8 percent. These results indicated that the impact of bypass leakage capacity on the containment pressure is significant. The bypass leakage capacity is discussed above.

Negative Pressure Design Evaluation. ESBWR DCD Tier 2, Revision 3, Section 6.2.1.1.4, states that the MSLB will not result in unacceptable results, but it does not indicate if other LOCAs were evaluated to conclude that this is the limiting case. In RAI 6.2-11, the staff requested that GEH discuss how the limiting cases were identified for both the drywell and wetwell. In response, GEH provided results of the inadvertent spray actuation analysis. The conclusion was that FWLB and MSLB scenarios are bounding possible containment conditions, with FWLB producing the highest peak drywell pressure and MSLB producing the lowest one, during the initial 2,000 seconds after the break. GEH modified DCD Tier 2, Section 6.2.1.1.4 accordingly.

RAI 6.2-11 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD Tier 2, Revision 5. Based on the applicant’s response, RAI 6.2-11 is resolved.

6.2.1.1.4 Conclusions

Based on the staff’s review of the submitted containment analysis, as presented in DCD Tier 2, Revision 9, and the staff’s independent confirmatory calculations of containment responses to the postulated DBA LOCAs, the staff finds the GEH containment analysis acceptable.

6.2.1.2 *Containment Subcompartments*

6.2.1.2.1 Regulatory Criteria

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 6.2.1.2, in accordance with SRP Section 6.2.1.2, Revision 3.

The acceptance criteria given below apply to the design and functional capability of subcompartments in the primary containment:

- GDC 4 as it relates to the environmental and missile protection provided to ensure that SSCs important to safety are designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during plant normal operations or during an accident

- GDC 50 as it relates to the subcompartments being designed with sufficient margin to prevent fracture of the structure because of pressure differential across the walls of the subcompartment

When performing analyses to demonstrate compliance with the requirements of GDC 50, the following assumptions and modeling schemes should be used to ensure that the results are conservative:

- The initial atmospheric conditions within a subcompartment should be selected to maximize the resultant differential pressure.

- Subcompartment nodalization schemes should be chosen such that there is no substantial pressure gradient within a node (i.e., the nodalization scheme should be verified by a sensitivity study that includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes). The guideline of Section 3.2 of NUREG–0609, "Asymmetric Blowdown Loads on PWR Primary Systems," issued January 1981, should be followed, and a nodalization sensitivity study should be performed, which includes consideration of spatial pressure variation (e.g., pressure variations circumferentially, axially, and radially within the subcompartment), for use in calculating the transient forces and moments acting on components.

- If vent flow paths are used that are not immediately available at the time of pipe rupture, the following criteria apply:
  - The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.
  - The validity of the analysis should be supported by experimental data or a testing program should be proposed at the construction permit or design certification stage that will support this analysis.
  - In meeting the requirements of GDC 4, the effects of missiles that may be generated during the transient should be considered in the safety analysis.

- The vent flow behavior through all flow paths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100-percent water entrainment.

In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the "frictionless Moody" (see Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Volume 87, page 134, February 1965) with a multiplier of 0.6 for water-steam mixtures and the thermal homogeneous equilibrium model for air-steam-water mixtures.

6.2.1.2.2 Summary of Technical Information

The design of the containment subcompartments was based on a postulated DBA occurring in each subcompartment.

For each containment subcompartment in which high-energy lines are routed, mass and energy release data corresponding to a postulated double-ended line break were calculated. The mass and energy release data, subcompartment free volumes, vent path geometry, and vent loss coefficients were used as input to an analysis to obtain the pressure/temperature transient response for each subcompartment. In addition to the drywell and the wetwell, the containment has two subcompartments, the drywell head region and the reactor shield annulus (RSA).

Drywell Head Region

The drywell head region is covered with a removable steel head, which forms part of the containment boundary. The drywell bulkhead connects the containment vessel flange to the containment and represents the interface between the drywell head region and the drywell. No high-energy lines are in the drywell head region.

Reactor Shield Annulus

The RSA exists between the RSW and the RPV. The RSW is a steel cylinder surrounding the RPV and extending close to the drywell top slab, as shown in DCD Tier 2, Revision 9, Figure 6.2-1. The opening between the RSW and the drywell top slab provides the vent pathway necessary to limit pressurization of the annulus resulting from a high-energy pipe rupture inside the annulus region. The shield wall is supported by the reactor support structure. Several high-energy lines extend from the RPV through the RSW. There are also penetrations in the RSW for other piping, vents, and instrumentation lines. The RSW is designed for transient pressure loading conditions from the worst high-energy line rupture inside the annulus region. GEH used the TRACG computer program to perform the RSA subcompartment evaluation.

6.2.1.2.3 Staff Evaluation

The staff reviewed DCD Tier 2, Section 6.2.1.2, and performed independent confirmatory analyses of the containment subcompartment by using alternative methodology (TRAC/RELAP Advanced Computational Engine (TRACE) computer code). The confirmatory calculations were based on additional information the staff requested in RAI 6.2-13, including synopsis of the piping break analyses, justification for the selection of the DBA (break size and location), and whether the leak-before-break was assumed to limit the pipe break area. In response, GEH stated that RSA was the only subcompartment, in addition to drywell and wetwell subcompartments, requiring assessment of pipe breaks. GEH assessed four types of pipe break (the MSL, FWL, and GDCS injection line and the bottom drain line) for the drywell and wetwell compartments. GEH assessed two types of pipe break (FWL and RWCU) for the RSW.

GEH selected the break locations to maximize the mass and energy released into the subcompartment. The break locations are usually the pipe segments on any flow path with the largest cross-section in the containment. GEH did not assume leak-before-break to limit the break area because it postulated DEG breaks for all pipe breaks.

RAI 6.2-13 was being tracked as a confirmatory item in the SER with open items. The staff reviewed the applicant's subcompartment analysis and, based on the staff's confirmatory calculations, accepted the results. The staff confirmed that the changes were incorporated in DCD Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-13 is resolved.

DCD Tier 2 was unclear as to whether pipe restraints are used to limit the break area of the pipe ruptures. Therefore, in RAI 6.2-14, the staff asked GEH to clarify. RAI 6.2-14 was being tracked as an open item in the SER with open items. In response, GEH stated that it took no credit in the analysis to limit the break area because of the presence of pipe restraints. The staff agreed with the information provided in the response. Based on the applicant's response, RAI 6.2-14 is resolved.

DCD Tier 2, Revision 2, Section 6.2.1.2, stated that a factor of 1.4 is applied to the peak differential pressure calculated for the subcompartment, structure, and the enclosed components. However, DCD Tier 2, Revision 2, Section 6.2.1.2.1, states that at least 15-percent margin above the analytically determined pressure is applied for structural analysis. Therefore, in RAI 6.2-15, the staff requested that GEH clarify. RAI 6.2-15 was being tracked as an open item in the SER with open items.

In response, GEH clarified that it is "at least 15 percent margin" applied for design-basis structural analysis. Also, in DCD Tier 2, Revision 5, the factor of 1.4 was changed to 1.2. The staff finds that the clarification and modifications are consistent with SRP Section 6.2.1.2 and acceptable. This addressed the staff's concern. RAI 6.2-15 is resolved.

The staff was unable to determine from the information provided in ESBWR DCD Tier 2 whether possible insulation collapsing in the containment subcompartment affects the vent areas used in the analyses. Therefore, in RAI 6.2-17, the staff requested that GEH clarify. In response, GEH stated that the RSA subcompartment vent areas in ESBWR containment are always open, and no insulation collapse would occur in this subcompartment. The staff finds that this response is acceptable as it provides design basis. In RAI 6.2-17 S01 the staff asked GEH to provide this information in the DCD. RAI 6.2-17 S01 was being tracked as an open item in the SER with open items. In response, GEH incorporated the above in DCD Tier 2, Revision 5. This addressed the staff's concern. RAI 6.2-17 S01 is resolved.

DCD Tier 2, Section 6.2.1.2.3, stated that the mass release rates are determined with Moody's frictionless critical flow model. This section also states that, when analyzed with TRACG, the peak subcompartment pressure responses were found to be below the design pressure for all postulated pipe break accidents.

DCD Tier 2, Section 6.2.1.2.3, stated that the TRACG computer code was used for the ESBWR containment subcompartment analysis. However, ESBWR DCD Tier 2 did not provide information on the conservatism of the blowdown model with respect to the pressure response of the subcompartment and a justification for using TRACG for subcompartment analysis. Therefore, in RAI 6.2-19 S01, the staff requested that GEH provide this information.

In response GEH stated that TRACG was qualified for analysis of the SBWR and ESBWR reactor system and containment in NEDC-32725P, TRACG Qualification for SBWR," Revision 1, August 2002, and NEDC-33083P. GEH provided results of time-step sensitivity analyses on peak maximum pressures and provided the sizes of the smallest nodes that are located around the postulated break. GEH agreed to provide this information in a proprietary licensing report for reference in the DCD. GEH stated that it had performed sensitivity studies to assess the effects of annulus volume, RSW vent flow area, and annulus hydraulic diameters and found the effects to be minor. After reviewing the GEH response, the staff finds that the GEH response addressed its concerns. GEH included its response in the revised licensing topical report (LTR), NEDE-33440P, Revision 1, "ESBWR Safety Analysis—Additional Information," issued June 2009. RAI 6.2-19 S01 is resolved.

ESBWR DCD Tier 2, Revision 1, did not provide the assumed initial operating conditions of the plant such as reactor power level and subcompartment pressure, temperature, and humidity which were assumed for the RSA subcompartment evaluation. Therefore, in RAI 6.2-20, the staff asked GEH to provide this information in the DCD. RAI 6.2-20 S01 was being tracked as an open item in the SER with open items. In response, GEH updated the DCD Tier 2, Section 6.2.1.2.3 in Revision 2 of the DCD to state that the containment subcompartment analysis assumed that the reactor is operating at full power and the containment is filled with dry air at atmospheric pressure and 100 degrees C (212 degrees F) when the postulated pipe break occurs. However, ESBWR DCD Tier 2 does not state whether the reactor power was adjusted to account for measurement uncertainties and does not justify using air while the ESBWR containment is inerted with nitrogen. Therefore, in RAI 6.2-20 S01, the staff asked GEH to clarify. In response, GEH stated that uncertainties associated with either "100% vs 102% power" or "air vs nitrogen" are bounded by the use of a 1.2 multiplier applied to the peak pressures calculated for annulus pressurization before being applied to the structural analyses. Based on its own independent analysis, the staff agrees with this information. Based on the applicant's response, RAI 6.2-20 S01 is resolved.

ESBWR DCD Tier 2 did not describe and justify the subsonic and sonic flow models used in vent flow calculations and did not state and justify the degree of entrainment assumed for the vent mixture. The staff needed this information to evaluate the ESBWR subcompartment loading. Therefore, in RAI 6.2-21, the staff requested that GEH provide this information. In response, GEH stated that it used the frictionless Moody critical mass flux correlation to model the break flow and that the model assumed critical velocity at the break and therefore was conservative. GEH stated that the degree of entrainment was not a TRACG input and it used the TRACG interfacial shear model described in the paper cited above by F.J. Moody. GEH revised DCD Tier 2, Revision 5, Section 6.2.1.2.3, accordingly. The staff finds that the GEH modeling of vent flow and entrainment is acceptable because it is consistent with SRP Section 6.2.1.2. This addressed staff's concerns. RAI 6.2-21 is resolved.

ESBWR DCD Tier 2 did not provide information on the containment subcompartment nodalization. Therefore, in RAI 6.2-23, the staff asked GEH to provide this information. In response, GEH provided nodal data but stated without specifics that it calculated large pipe and vessel support structure volumes and hydraulic diameters and accounted for the additional obstructions by applying a 10-percent reduction factor in the annulus volume for cells where a specific obstruction is not modeled. The staff needed the details of nodalization to perform its confirmatory analysis, and staff requested this information in RAIs 6.2-23 S01-S03. RAIs 6.2-23 S01-S03 were tracked as open items in the SER with open items.

In response to RAIs 6.2-23 S01-S03, GEH provided the requested information. The staff confirmed that the discussion addressing these concerns is included in NEDE-33440P, Revision 1.

The staff reviewed and accepts the applicant's response as it is consistent with previously approved Mark III methodology and also is supported by the insights gained from subcompartment analysis performed independently by the staff using alternate methodology (TRACE code). Based on the applicant's response, RAIs 6.2-23 S01-S03 are resolved.

ESBWR DCD Tier 2 did not provide graphs of the pressure responses of subnodes within a subcompartment as functions of time. This information is needed for evaluations of the effect on structures and component supports. Therefore, in RAI 6.2-24, the staff asked GEH to provide this information. In response, GEH provided graphs of the pressure responses of subnodes within a subcompartment as functions of time, which were acceptable because they addressed the staff's concern. In RAI 6.2-24 S01 the staff requested that GEH add this information to the DCD. RAIs 6.2-24 and 6.2-4 S01 were being tracked as open items in the SER with open items. In response, GEH provided the requested information in NEDE-33440P, Revision 1, which is referenced in the DCD. After reviewing the GEH responses the staff finds them acceptable. Based on the applicant's response, RAIs 6.2-24 and 6.2-24 S01 are resolved.

ESBWR DCD Tier 2 did not provide the mass and energy release data for the postulated pipe breaks. Therefore, in RAI 6.2-25 S01 the staff asked GEH to provide this information. In response, GEH provided the method used to calculate mass and energy release data but not the actual data. Therefore, in RAI 6.2-25 S01 the staff asked GEH to provide this information and update DCD Tier 2. In response, GEH provided the requested information in NEDE-33440P, Revision 1, which is referenced in the DCD. RAI 6.2-25 S01 is resolved.

ESBWR DCD Tier 2 did not state the flow conditions (subsonic or sonic) for vent flow paths up to the time of peak pressure. The staff needs this information to evaluate ESBWR subcompartment loads per SRP Section 6.2.1.2 and RG 1.70, Section 6.2.1.2. Therefore, in RAI 6.2-26, the staff asked GEH to provide this information. In response, GEH stated that before the time of peak pressure, the vent flow is subsonic. GEH agreed to update the DCD to include this information. RAI 6.2-26 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the information was incorporated in DCD Tier 2, Revision 5, Section 6.2.1.2.3. Based on the applicant's response, RAI 6.2-26 is resolved.

In RAI 6.2-29, the staff expressed concern about the GEH methodology, specifically, with applying the TRACG computer program to the containment subcompartment analysis without providing information on the time-step and nodalization study, code validation, and comparison to approved methods. In response to RAIs 6.2-29 S01-S03 GEH provided a comparison of the TRACG and CONTAIN analyses.

RAIs 6.2-29 S01-S03 were tracked as open items in the SER with open items. Based on the submitted additional comparison and the staff's own confirmatory analysis performed with a subcompartment code TRACE, the staff accepts the results of the GEH subcompartment analysis. Based on the applicant's response, RAIs 6.2-29 S01-S03 are resolved.

6.2.1.2.4 Conclusions

The staff reviewed the application of the TRACG computer program to the subcompartment analysis and its comparison to alternative methodology (the CONTAIN code). Based on the review, and the staff's own independent analysis (with the TRACE code), the staff finds the GEH subcompartment analysis to be sufficiently conservative and, therefore, acceptable.

6.2.1.3 *Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents*

Section 6.2.1.1 of this report presents the staff's review of the DCD to determine if it meets the criteria of SRP Section 6.2.1.3.

6.2.1.4 *Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures*

SRP Section 6.2.1.4, applies to PWRs and thus is not applicable to the ESBWR.

6.2.1.5 *Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies*

SRP Section 6.2.1.5, applies to PWRs and thus is not generally applicable to the ESBWR. However, during a December 2006 audit, the staff raised the issue of possible implications of the minimum containment pressure on the initiation timing of GDCS injection and thus on ECCS performance. As described in Section 6.2.1.1 of this report, this issue was resolved by issuing RAI 6.2-144. GEH added DCD Tier 2, Revision 4, Appendix 6C, to provide an evaluation of the impact of containment backpressure on the ECCS performance.

6.2.1.6 *Suppression Pool Dynamics Loads*

6.2.1.6.1 Regulatory Criteria

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 6.2.1.6, in accordance with SRP Section 6.2.1.1.C, Revision 7, issued March 2007. To meet the requirement of GDC 4, regarding the dynamic effects associated with normal and accident conditions, calculation of dynamic loads should be based on appropriate analytical models and supported by applicable test data. The calculations should consider loads on suppression pool retaining structures and structures that may be located directly above the pool, as a result of pool motion during a LOCA or following actuation of one or more reactor coolant system SRVs.

6.2.1.6.2 Summary of Technical Information

GEH submitted DCD Tier 2, Appendix 3B, to define the containment hydrodynamic load definitions for the ESBWR. The methodology used to develop these load definitions and the justification for their applicability to the ESBWR is given in a proprietary report, NEDE-33261P, "ESBWR Containment Load Definition," issued May 2006.

NEDE-33261P provides a description and load definition methodology for hydrodynamic forces acting on the ESBWR primary containment during a postulated LOCA and/or SRV or DPV actuation. The load definition methodology used for the ESBWR containment design is similar to that used for earlier BWR containment designs and particularly similar to that used and approved for the ABWR design.

The geometries of the pressure suppression systems in the ABWR and ESBWR designs are similar. Table 6.2.1.6-1 of this report lists the key differences between the two containment designs.

Table 6.2.1.6-1. Geometries of the pressure suppression system.

Parameter	ESBWR	ABWR
Number of vertical vents	12	10
Suppression pool angular sector per vertical vent (degrees)	30	36
Pool depth (m)	5.5 (18.0 ft)	7.1 (23.3 ft)
Top vent submergence (m)	2.0 (6.6 ft)	3.6 (11.8 ft)
Distance from vent exit to outer containment wall (m)	9.0 (29.5 ft)	6.85 (22.47 ft)
Pool surface area per vent (m^2)	66.6 (716.9 ft^2)	50.7 (545.7 ft^2)
Vertical vent distance between drywell entrance and top vent entrance (m)	9.35 (30.6 ft)	17.0 (55.8 ft)

In both the ABWR and ESBWR designs, the drywell and the annular suppression pool are connected by a set of circular vertical vents of the same diameter, each with three circular horizontal vents, also of the same diameter, and at the same elevations, extended into the suppression pool to the same distance.

Since there is a high degree of geometric similarity between the ESBWR and ABWR containments, the physical phenomena associated with the postulated DBA events during the first few minutes into the accidents are identical for both designs. The following is a description of these phenomena, based on NUREG–1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," issued July 1994, and NEDE-33261P.

LOCAs and SRV discharges, as well as the DPV actuation, are the events that can impose dynamic loads on the suppression pool. SRVs discharge steam from the reactor pressure vessel through discharge piping that is routed into the suppression pool and fitted at the pool end with a quencher to enhance heat transfer between the hotter SRV discharge fluid (steam and air) and the cooler suppression pool water. The DPV discharges the mass and energy to the containment, increasing the mass flux through the main vents. However, this additional mass flux is bounded by the LOCA vent mass flux and, therefore, the containment hydrodynamic loads calculated for the DBA LOCA are used for the design.

Since the ESBWR design has no recirculation line, the largest postulated pipe breaks are FWLB and MSLB. The dynamic loads in the suppression pool caused by these events can be characterized by several phenomena that occur in the order of (1) vent clearing, (2) PS, (3) high steam flow, and (4) chugging (CH). After an FWLB or MSLB, with sufficient pressurization of the drywell, water in the vents is forced out into the pool. This vent water clearing causes submerged jet-induced loads on nearby structures and the pool basemat. After vent clearing, an air and steam bubble flows out of the vents. The air component, originating from the drywell, expands in the pool causing a rise in pool surface level, referred to as PS, and imposing loads on submerged structures and pool boundaries. After PS, a period of high steam flow occurs, and steam is condensed in the pool vent exit area, causing pressure oscillations in the pool. This phenomenon, referred to as condensation oscillation (CO), produces oscillatory and steady loadings on the containment structure. Later, as vent steam flow decreases, a steam bubble may occur, and its sudden collapse creates oscillatory loads. This process (CH) imposes significant vent and suppression pool boundary loads.

The CO experiments (e.g., NEDC-31393, Revision 0, “Containment Horizontal Vent Confirmatory Test, Part I” [proprietary]) indicate that the wall, liner, and submerged structures within two vent diameters of each horizontal vent also experience local effects. The methodology, as presented in NEDE-33261P, addresses this phenomenon.

One of the unique design features of the ESBWR is the PCCS (see Section 6.2.2). Its operation, which immediately follows a LOCA, would mitigate to some extent the PS loads calculated for the scenario described above, although the LOCA analysis did not credit the performance of the PCCS for the first several minutes of the postulated accident.

Other postulated LOCAs, intermediate and small, lead to similar scenarios and the resulting PS, CO, and CH loads are bounded by those calculated for the DBA LOCA.

For certain reactor transients, the pressure relief is through activation of the SRV. For these events, the steam discharge into the suppression pool consists of three phases: water clearing, air clearing, and steam flow. The discharge pipe standing column of water first is pushed out, or cleared, into the pool by blowdown steam pressure. Water clearing creates SRV pipe pressure and thermal loads, pipe reaction forces, drag loads on structures submerged in the pool, and pool boundary loads. After water clearing, air clearing occurs as air above the water column in the pipe is forced out of the pipe and into the pool. The air-clearing phase generates expanding bubbles in the pool that cause transient drag loads on a submerged structure as a result of both the velocity and acceleration fields and oscillating pressure loads on the pool boundary. Finally, the steam-flow phase creates pipe reaction forces, quencher thrust forces, structure thermal loads, and oscillating pool boundary loads as a result of steam jet condensation at the quencher.

The ESBWR SRV discharge is directed to the suppression pool through X-quenchers that GEH has stated are identical to the quenchers used for the Mark III designs. GEH also stated that the calculation methodology used for establishing the ESBWR quencher discharge loads is the same as previously used for ABWR, Mark II, and Mark III containments. In brief, the methodology is based on empirical correlations derived from the test of various scales. Therefore, GEH concluded that the hydrodynamic load methodology developed for the Mark II and Mark III designs was applicable to both the ESBWR suppression pool geometry and the X-quencher configuration.

During the ABWR review, the staff raised an issue concerning the SRV loads that would result from a second opening of the SRV while the SRV tailpipe is still hot from the initial discharge; the staff referred to this as “subsequent actuation” or “consecutive actuation” in NUREG–0802, “Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments,” issued October 1982. The concern was that a subsequent SRV actuation could generate higher loads on the structure. However, the subsequent actuation effect is considered in the methodology as described in NEDE-33261P. Therefore, the staff accepted the GEH position that the methodology GEH used to calculate hydrodynamic loading on SRV discharge piping resulting from the initial and subsequent SRV actuations is consistent with the methodology used for earlier BWR (Mark II and III) designs.

The ESBWR suppression pool configuration is similar to that of the ABWR, as shown in Table 6.2.1.6-2 of this report.

Table 6.2.1.6-2. Suppression pool configuration.

Design Feature	ESBWR	ABWR
Reactor power, MWt	4,500	4,000
Drywell volume, m^3 (ft^3)	7,206 (~254,520)	7,350 (~259,500)
Wetwell gas space volume, m^3 (ft^3)	5,350 (~188,900)	5,960 (~210,000)
Vertical vents (total), m^2 (ft^2)	13.6 (146)	11.6 (125)
Pool surface only, m^2 (ft^2)	799 (~8,600)	507 (~5,450)

Potentially, a slightly higher power and a slightly smaller drywell volume may increase the hydrodynamic forces. However, these negative effects are more than offset by a larger vent area, a larger pool volume, and a larger pool surface area.

Based on these similarities, GEH considers the methodology used to evaluate the pool response to a postulated accident (i.e., pool boundary loads resulting from bubble formation, the PS velocity and acceleration, the pool surface elevation, and the wetwell gas space pressure) for the ABWR design to be equally applicable to the ESBWR containment.

Adjustments for ESBWR Application

Although the ESBWR and ABWR pressure suppression systems are similar, there are some differences in specific dimensions. These differences were accounted for as described below.

For PS, the methodology approved for the ABWR required no adjustment. One difference is that there are no vacuum breakers or upward diaphragm loads since, during the PS phase (0 to 5 seconds), the wetwell pressure is always lower than the drywell pressure. As this conclusion is based on analyses for the six postulated cases, it needs to be demonstrated under inspection, test, analysis, and acceptance criteria (ITAAC) 1 and 8 in DCD Tier 1, Table 2.1.1-3.

For CO loads, an additional pressure time history was added by compressing the time scale of the time history with the highest frequency content. The frequency was increased by the ratio of ESBWR-to-ABWR vertical distance from the vent entrance to the top vent (approximately 1/1.8). This additional pressure signature is to account for any possible influence of vent acoustic modes on the CO frequency.

For CH loads, to adjust the ABWR CH frequency to the ESBWR, the frequency was increased by the ratio of ESBWR-to-ABWR pool depth ratio (approximately 1/1.3).

For both CO and CH loads, the pressure amplitude was increased by a factor of 1.2. Although, given the ESBWR pool geometry, this additional conservatism is not necessary, it is included as part of the initial design assumptions.

For SRV loads, the X-quencher methodology, as described and reviewed in NUREG–1503, is used without adjustment.

Effect of Unique ESBWR Features

The PCCS, described in Section 6.2.2 of this report, receives a steam-gas mixture directly from the drywell. Most, if not all, steam is condensed in the tubes, and the remaining gas, primarily noncondensables, is deposited in the suppression pool. These PCCS characteristics reduce the CO loads and prevent the occurrence of the CH loads. In addition, the small venting area and low submergence of the vent line minimize the effect of PS, bounded by the LOCA loads.

The GDCS pools, described in Section 6.3 of this report, are equipped with spillover pipes to direct potential water overflow to the entrance of the main vents. In Revision 1 of NEDE-33261P, GE stated that these pipes have no impact on containment thermal-hydraulic loads.

The largest postulated pipe breaks in the ESBWR are FWLB and MSLB since there is no recirculation line. Because of more rapid pressurization during the MSLB, the MSLB loads bound the FWLB PS loads. For CO and CH loads, both breaks need to be evaluated. The review of thermal-hydraulic conditions revealed that the predicted steam mass fluxes for the ESBWR MSLB and FWLB are well below the values measured during the horizontal vent tests used for the ABWR load definition. Therefore, the ABWR CO and CH load definitions are applicable to the ESBWR design.

The ESBWR pool-to-vent area ratio is about 58; for the ABWR, the ratio is about 40; for Mark II, the ratio is typically 20.0; and for Mark III, it is typically 12.0. GEH believes that the larger pool relative to the vent area will cause a reduction in the pool hydrodynamic loads. NUREG–0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria," issued August 1981, supports this position.

The shallower and wider ESBWR pool and two additional vents tend to produce lower pressure amplitude, while a lower mass flow rate produces frequencies in the lower range of the existing experimental database.

6.2.1.6.3 Staff Evaluation

The staff considered the differences between the Mark II, Mark III, and ABWR databases in determining whether the ESBWR suppression pool wall pressures do not exhibit any unusual characteristics when compared to the Mark III wall pressures. Because the ABWR and ESBWR suppression pool designs are so similar, the staff reviewed a concern (described in NUREG–1503) regarding the scaling loads used by GEH for developing the load definition. The ABWR-specific subscale (SS) and partial full-scale (FS) tests appear to be adequate representations of the ESBWR main vents for predicting the suppression pool hydrodynamic response for unstable CO and CH loads. However, DCD Tier 2, Revision 3, did not discuss the applicability of the SS and FS tests to the ESBWR design. (The SS facility has a single horizontal pipe, and the FS facility has two horizontal pipes, while the ESBWR has three horizontal vent pipes extended into the suppression pool.) Also, the staff expressed concerns about the Mark III data from the pressure suppression test facility blowdown tests, reported in NUREG–0978, "Mark III LOCA-

Related Hydrodynamic Load Definition," issued August 1984, which were conducted with FS vent lengths and all three horizontal vents. In RAI 6.2-158, the staff asked GEH to address the above issues. In response, GEH referred to the revised "ESBWR Containment Load Definition" report (NEDE-33261P, Revision 1,) which addressed the staff's concerns. The report demonstrated that the ABWR CO wall load definition was based on SS tests, and the ABWR CH load definition was based on FS tests. RAI 6.2-158 was being tracked as an open item in the SER with open items. The staff accepted these load definitions during the ABWR certification process. Since the similarity between the ABWR and ESBWR containment systems was established, the staff finds the response acceptable. Based on the applicant's response, RAI 6.2-158 is resolved.

As currently implemented in the Mark I, II, and III designs, the suppression pool temperature limits involve a three-tier approach. The lowest temperature threshold requires the operator to take actions such as activating pool cooling to reduce the suppression pool temperature. The plant, however, can continue to operate at power during this time. The intent of this threshold is to ensure that the operator acts to reduce pool temperature. This temperature is typically 35 degrees C (95 degrees F). Operation can continue until the suppression pool reaches 43 degrees C (110 degrees F). At this temperature, an automatic scram on high suppression pool temperature occurs. Finally, if the pool reaches 49 degrees C (120 degrees F), the TS require depressurization of the reactor coolant system and initiation of cold shutdown conditions. The ESBWR TS 3.6.2.1, "Suppression Pool Average Temperature," specifies temperature thresholds for reactor scram, shutdown, and vessel depressurization of 43 degrees C, 49 degrees C, and 54 degrees C (110 degrees F, 120 degrees F, and 130 degrees F), respectively. These limits do not follow the guidance provided in NUREG–0783, "Suppression Pool Temperature Limits for BWR Containment," issued November 1981. In RAI 6.2-159 the staff asked for explanation why the NUREG–0783 guidance was not followed including a description of the effect of pool temperature on the SRV load evaluation. RAI 6.2-159 was being tracked as an open item in the SER with open items. In response, GEH stated that additional test data with X-Quencher, used in the ESBWR, collected after NUREG–0783 was issued, justified elimination of the local pool temperature limit. The staff approved this conclusion in a letter from G. Holahan (NRC) to R. Pinelli (Boiling Water Reactor Owners Group), dated August 29, 1994. The separate but related issue of potential steam ingestion into ECCS pump suction does not apply to the passive ESBWR design. In addition, the TS pool temperature limit requirement is consistent with the assumptions used for the ESBWR safety analyses. Based on the applicant's response, RAI 6.2-159 is resolved.

NEDE-33261P (May 2006) Revision 0, implies that GEH used the PICSM computer code to compare Mark III suppression PS test data from the pressure suppression test facility with analytical predictions. GE technical report NEDE-21544P, Revision 0, "Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon" (proprietary), issued December 1976, describes the code. GE validated the test data generated for the Mark II design; however, the staff did not review and approve the code. GEH addressed the staff's concern with potential liquid and froth impacts on the vacuum breaker valves in its response to RAI 6.2-160. RAI 6.2-160 was being tracked as an open item in the SER with open items. DCD Tier 2, Revision 5, requires the diaphragm floor slab to be greater than 9,600 mm (31.5 ft) above the wetwell floor. This requirement ensures that the maximum PS of 4,100 mm (13.5 ft) will not reach the vacuum breaker valves assuming the maximum allowable pool depth of 5,500 mm (18.0 ft), as specified by TS SR 3.6.2.2.1. The froth impacts are predicted by using the same methodology as previously approved for the ABWR certification. Based on the established similarity between the ABWR and ESBWR containments, the staff accepts the response. Based on the applicant's response, RAI 6.2-160 is resolved.

GEH applied the Mark II hydrodynamic loads to the ESBWR design. The staff documented its evaluation of the definition of the Mark II design containment hydrodynamic load in NUREG–0808. In the evaluation of the PS phenomena (discussed in Section 2.1 of NUREG–0808), the staff relied on comparisons and a substantial amount of data from tests conducted by both GEH and the Japan Atomic Energy Research Institute. These tests were directly applicable to the Mark II design. GEH developed a computer program PSAM (described in NEDO-21061, Revision 0, “Mark II Containment Dynamic Forcing Functions Information Report” issued September 1975) to be used as part of the Mark II hydrodynamic load evaluation program. The staff has reviewed the Mark II program and approved the methodology and PSAM in NUREG–0808. However, it did not find the GEH methodology within PSAM acceptable. Rather, the staff based its acceptance on the favorable comparisons with the database. In RAI 6.2-161, the staff requested that GEH address the above issue. RAI 6.2-161 was being tracked as an open item in the SER with open items. In response, GEH explained that both the Mark II program and the ABWR certification used a different computer program, PICSM, for the pool hydrodynamic loads, as presented and approved by the staff in NUREG–1503. Based on design similarities between the ABWR and ESBWR designs, as discussed in NEDE-33261P, Revision 1, GEH claimed that this methodology can be applied to the ESBWR hydrodynamic loads definition. Based on the use of methodology previously accepted during the ABWR certification process and the established similarity between the ABWR and ESBWR containments, the staff accepts the applicant’s response. Based on the applicant’s response, RAI 6.2-161 is resolved.

In RAI 6.2-164, the staff requested details of analysis of the suppression pool and its associated structure, systems, and components (SSCs) subjected to hydrodynamic loads as described in DCD Tier 2, Revision 4, Appendix 3B. In response the applicant added Appendices 3F and 3G to DCD Tier 2, Revision 4 providing qualification for the suppression pool and its associated SSCs to withstand imposed hydrodynamic loads. GEH addressed an additional concern regarding the integrity of the PCCS vent pipe (described by the staff in RAI 6.2-164 S01), in its response to RAI 14.3-131 S01 wherein GEH indicated that the PCCS piping is included in the ITAAC in DCD Tier 1, Section 3.1. RAI 6.2-164 S01 was being tracked as an open item in the SER with open items. Based on the review of the Appendices 3F, 3G, and the audits performed on the applicant's suppression pool analyses, the staff finds the GEH responses regarding the above concerns to be acceptable. Based on the applicant’s response, RAI 6.2-164 S01 is resolved.

6.2.1.6.4 Conclusions

The staff reviewed the methodology presented in NEDE-33261P, including Revision 1, and used for evaluation of the ESBWR hydrodynamic loads. The analytical models of the involved physical phenomena are the same as those used for the safety evaluation of the approved ABWR design. The review included evaluation of the applicability of the rationale the staff used in the ABWR design approval process. Also, the staff reviewed the relevant database from previous BWR research programs.

In a separate evaluation, the staff reviewed and approved the application of the TRACG code for the ESBWR pool dynamic analysis (letter from W.D. Beckner (NRC) to L. Quintana (General Electric Nuclear Energy [GENE]), “Safety Evaluation Report Regarding the Application of GENE’s TRACG Code to ESBWR LOCA Analyses,” dated August 19, 2004). The staff also acknowledges that, compared to the approved ABWR design, the shallower and wider ESBWR pool and the two additional vents tend to produce lower pressure amplitude, while a lower mass flow rate produces frequencies in the lower range of the existing experimental database. Therefore, the staff finds the methodology presented in NEDE-33261P to be acceptable.

6.2.1.7 *Containment Debris Protection for Emergency Core Cooling System Strainers*

6.2.1.7.1 Regulatory Criteria

SRP 6.2.2, Revision 5, states that to satisfy the requirements of GDC 38 and 10 CFR 50.46(b)(5) regarding the long-term spray system(s) and ECCS(s), suppression pools in BWRs should be designed to provide a reliable, long-term water source for ECCS and containment spray system pumps.

RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, November 2003, as supplemented by the NRC-approved Boiling Water Reactor Owners' Group Utility Resolution Guidance, provide guidance for BWR debris evaluations.

The following NRC bulletins (BLs) provide additional guidance:

- Bulletin (BL) 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993.
- BL 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994.
- BL 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995.
- BL 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996.
- BL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated July 14, 1998.

6.2.1.7.2 Summary of Technical Information

ESBWR DCD Tier 2, Revision 9, Section 6.3.2.7.2, states that suppression pool equalization lines have an intake screen to prevent the entry of debris material into the system that might be carried into the pool during a large-break LOCA. A perforated steel plate will cover the GDCS pool airspace opening to the drywell to prevent debris from entering the pool and potentially blocking the coolant flow through the fuel. The maximum hole diameters in the perforated steel plate are 38 mm (1.5 inch).

6.2.1.7.3 Staff Evaluation

The ESBWR GDCS or PCCS does not have active pumps that are required for core cooling or containment heat removal during the 72 hours and beyond following a design-basis LOCA. The staff reviewed the DCD to determine that latent or LOCA-generated debris will not clog the GDCS or PCCS flow paths.

DCD Tier 2, Revision 1, Section 6.2.1.1.2, states the following:

> There is sufficient water volume in the suppression pool to provide adequate submergence over the top of the upper row of horizontal vents, as well as the PCCS return vent, when water level in reactor pressure vessel (RPV) reaches one meter above the top of active fuel and water is removed from the pool during post-loss-of-coolant accident (LOCA) equalization of pressure between RPV and the wetwell. Water inventory, including the GDCS, is sufficient to flood the RPV to at least 1 m above the top of active fuel.

The DCD was not clear as to how water is removed from the suppression pool during the post-LOCA period. Therefore, in RAI 6.2-6, the staff asked GEH for clarification. In response, GEH stated that during the post-LOCA period, the suppression pool equalization line will open, allowing water to flow from the suppression pool to the RPV.

If the ESBWR design relies on the suppression pool equalization line to maintain a depth of 1 m (3.28 ft) of water above active fuel in the RPV, the suppression pool equalization line should be designed as such. To review the functioning of the suppression pool equalization line during DBA LOCA scenario, in RAI 6.3-40, the staff requested GEH to provide the value of differential pressure across the equalization line check valves for each of the DBA LOCA scenario analyzed.

In response, GEH stated that the suppression pool equalization line will not open for 72 hours and beyond for all design-basis LOCA scenarios. DCD Tier 2, Revision 3, Section 6.3.2.7.2, states that "[s]uppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA." The ESBWR DCD was not clear as to how the intake strainer is designed to prevent the entry of debris material into the system. Therefore, in RAI 6.2-6 S01, the staff asked GEH to explain. RAI 6.2-6 S01 and RAI 6.3-40 were being tracked as open items in the SER with open items.

In response to a related RAI, RAI 6.2-173 S01 which is described below in this section, GEH stated the following:

> As stated in the response to RAI 6.3-40 (MFN 06-488, dated December 22, 2006), and confirmed in the response to RAI 6.3-40 S01 (MFN 06-488, Supplement 1, dated December 21, 2007), reactor pressure vessel (RPV) water levels stay above Level 0.5 setpoint for 72 hours and beyond for all loss-of-coolant accident (LOCA) scenarios. In addition, a 30 day analysis confirms RPV water level stays above Level 0.5 setpoint, as discussed in the response to RAI 6.2-140 S02 (MFN 08-633, dated August 18, 2008). Therefore, the squib valves in the GDCS equalization lines never open, and the GDCS equalization lines are not required to function in response to a LOCA and do not perform a safety-related function. Therefore, the application of Regulatory Guide (RG) 1.82, Revision 3, is not required.

After reviewing the GEH response to RAI 6.2-173 S01, the staff finds that an intake strainer for the suppression pool equalization line is not required for 30 days following a LOCA. This addresses the staff's concern raised in RAI 6.2-6 S01 and RAI 6.3-40. RAI 6.2-6 S01 and RAI 6.3-40 are resolved.

DCD Tier 2, Revision 1, Section 6.3.2.7.2, states that the GDCS pool airspace opening to the drywell will be covered by a mesh screen or the equivalent to prevent debris from entering the pool and potentially blocking the coolant flow through the fuel. Although a mesh screen could protect GDCS pools from the entrance of some debris, it will not stop debris smaller than the mesh size from entering. Debris that enters the GDCS pool could flow with the GDCS injection flow into the vessel and could potentially block the coolant flow through the fuel. Therefore, in RAI 6.3-41, the staff asked GEH to explain what action it would take to prevent such debris blockage. In response, GEH stated that it would use a perforated steel plate instead of a mesh screen to protect the GDCS pool from the entrance of debris and that the holes in the perforated steel plate will be smaller than the orifice holes in the fuel support castings. In RAI 6.3-41 S01 the staff requested the specific dimensions of the perforated plate holes, fuel assembly inlet orifice diameter, and the minimum GDCS line diameter. The staff needed this information to confirm that the holes in the perforated plate are small enough to prevent the entrance of debris that could block the fuel inlet orifice. In response, GEH provided the requested information, and agreed to add this information to the DCD.

DCD Tier 2, Revision 3, Section 6.3.2.7.2, states that the GDCS injection system consists of one 200-mm (8-in.) pipe mounted with a temporary strainer. The staff's concern was that debris could clog the temporary strainers and consequently impede the GDCS injection flow. Therefore, in RAI 6.3-41 S01 the staff asked GEH to explain the effect of the temporary strainer on the GDCS injection flow. In response, GEH stated that the temporary strainer was not intended to remain as part of the system configuration and that the strainer will be removed after initial flushing of the GDCS injection lines. GEH agreed to update the DCD to include this information. The staff finds that this response addresses its concerns and is acceptable. GEH needed to update the DCD to include the remaining information as described above. RAI 6.3-41 S01 was being tracked as an open item in the SER with open items.

GEH updated DCD Tier 2, Revision 4, Section 6.3.2.7.2, to provide the dimensions of the holes in the perforated plate and to state that the temporary strainer will be removed after initial flushing of GDCS injection lines. This addresses the staff's concerns raised in RAI 6.3-41 S01. RAI 6.3-41 S01 is resolved.

During a LOCA, if the PCCS heat exchanger inlets are within the zone of influence, debris ingress is expected. However, DCD Tier 2, Revision 2, did not address the impact of possible debris ingress into the PCCS. Therefore, in RAI 6.3-42, the staff requested that GEH describe the impact of the debris on the performance of the heat exchanger. In response, GEH stated that the PCCS heat exchanger inlet pipe is provided with a debris filter with holes no greater than 25 mm (1 in.) to prevent entrance of missiles into the pipe and protection from fluid jets during a LOCA. These holes are smaller than the size of the heat exchanger tubes (50-mm (2-in.) nominal diameter), which have the smallest diameter of the piping components in the PCCS. GEH stated that if there is any debris that enters the PCCS, it cannot become lodged in the vertical heat exchanger tubes where the heat transfer function is performed, and thus, debris will not impact the PCCS performance. The staff finds that the PCC inlet pipe debris filter would limit debris entering the PCCS during a LOCA and that the PCCS heat transfer function would not be impacted. This addressed the staff's concern. In RAI 6.3-42 S01 the staff requested that the dimension of the holes of the debris filter should be added to the DCD. RAI 6.3-42 S01 was being tracked as an open item in the SER with open items. GEH revised DCD Tier 2, Revision 5, Section 6.2.2.2.2, to include this dimension. RAI 6.3-42 S01 is resolved.

The ESBWR relies on the PCCS to provide water to the GDCS for core cooling and for containment heat removal for 72 hours after a LOCA. Beyond 72 hours, the ESBWR also relies on the FAPCS. DCD Tier 2, Revision 3, Table 19A-2, identifies the FAPCS operating in suppression pool cooling and LPCI modes as being subject to RTNSS.

However, DCD Tier 2, Revision 3, Table 1C-1, states that BL 95-02 is not applicable to the ESBWR because it does not have a safety-related suppression pool cooling system. The same table states that BL 93-02 and its Supplement 1, BL 96-03, and BL 98-04 do not apply to the ESBWR because the reactor design provides emergency core cooling via the GDCS and the GDCS pools do not have the debris transport mechanisms to which the suppression pool is subject.

Therefore, in RAI 6.2-173, the staff requested that GEH explain why the debris-plugging issues described in the above BLs should not be applied to the debris plugging of the suppression pool suction strainer for operation of the FAPCS 72 hours after a LOCA. RAI 6.2-173 was being tracked as an open item in the SER with open items.

In its response to RAI 6.2-173, GEH stated the following:

> Long-term decay heat removal from the containment is provided by the Passive Containment Cooling System (PCCS), and after 72 hours the PCCS vent fans are available to increase the efficiency of the PCCS condensers. The PCCS along with the vent fans are capable of maintaining containment pressure below the design pressure for 30 days as described in the response to RAI 6.2-140 S02. In addition, the FAPCS lines associated with the suppression pool are not considered to be operational during a LOCA event and would not be considered available for operation until the seventh day after the start of a LOCA event. Therefore, only when determined to be appropriate and available, the FAPCS may be actuated in the low pressure coolant injection (LPCI), suppression pool cooling, or drywell (DW) spray modes to provide additional cooling to bring the plant to cold shutdown. Since the long term operation of the PCCS vent fans is sufficient to protect the integrity of containment, function of the FAPCS suppression pool line is not safety-related and the operation of the FAPCS cooling function is not required. Therefore RG 1.82, Revision 3 is not applicable to this application.

After reviewing the GEH response, the staff determined that RG 1.82, Revision 3, is not applicable to the ESBWR because the FAPCS cooling function is not required and the PCCS and the vent fans are capable of maintaining containment pressure below the design pressure for 30 days. This addresses the staff's concerns raised in RAI 6.2-173. RAI 6.2-173 is resolved.

6.2.1.7.4 Conclusions

The staff finds that the ESBWR design includes features to limit debris affecting the performance of the decay heat removal function following a LOCA. The staff determined that RG 1.82, Revision 3, is not applicable to the ESBWR because the FAPCS cooling function is not required and the PCCS and the vent fans are capable of maintaining containment pressure below the design pressure for 30 days. The staff finds the ESBWR design acceptable because LOCA-generated or latent debris will not affect the ability of the ESBWR design to meet GDC 35, 38, and 41.

6.2.2 Containment Heat Removal System

6.2.2.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 6.2.2, in accordance with SRP Section 6.2.2, Revision 5. The applicant's containment heat removal system is acceptable if it meets the requirements of the following Commission regulations:

- GDC 38, as it relates to the following:
  - 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
  - The ability of the containment heat removal system to perform in a manner consistent with the function of other systems
  - The safety-grade design of the containment heat removal system providing suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capability to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished in the event of a single failure

- GDC 39, "Inspection of containment heat removal system," as it relates to the design of the containment heat removal system to permit periodic inspection of components

- GDC 40, "Testing of containment heat removal system," as it relates to (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system

The regulations governing the evaluation of standard plant designs explicitly recognize the unique characteristics of the ESBWR PCCS. The regulation in 10 CFR 52.47(b)(2)(i)(A) states that, in the absence of a prototype plant that has been tested over an appropriate range of normal, transient, and accident conditions, a plant that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions" must meet the following requirements:

- The performance of each safety feature of the design has been demonstrated either through analysis, appropriate test programs, experience, or a combination thereof.

- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

- Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

6.2.2.2 *Summary of Technical Information*

Consistent with the applicable requirements, GEH, developed and performed design certification tests of sufficient scope, including both separate effects and integral systems experiments, to provide data with which to assess the computer programs used to analyze plant behavior over the range of conditions described in the third requirement above. To satisfy the requirements of 10 CFR 52.47(c)(2)(i)(A), GEH developed test programs to investigate the PCCS, including both component and phenomenological (separate effects) tests and integral systems tests.

The PCCS removes the core decay heat rejected to the containment after a LOCA. It provides containment cooling for a minimum of 72 hours post-LOCA, with containment pressure never exceeding its DPL, and with the IC/PCC pool inventory not being replenished.

GEH considers the PCCS condenser as an extension of the containment pressure boundary, and the PCCS condenser is used to mitigate the consequences of an accident. This function classifies it as a safety-related ESF. ASME Code, Section III, Class 2, and Section XI requirements for design and accessibility of welds for inservice inspection (ISI) apply to meet GDC 16. Quality Group B requirements apply as described in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3, February 1976. The system is designed to seismic Category I per RG 1.29, "Seismic Design Classification," Revision 4. The common cooling pool shared by the PCCS condensers and the ICs is a safety-related ESF, and it is designed such that no locally generated force (such as an ICS rupture) can destroy its function. Protection requirements against mechanical damage, fire, and flood apply to the common IC/PCC pool.

The PCCS condenser is sized to maintain the containment pressure within its design limit for DBAs. DCD Tier 2, Revision 9, Section 6.2.2.2.2 states, “The system is designed as a passive system with no components that must actively function in the first 72 hours after a DBA, and it is also designed for conditions that equal or exceed the upper limits of containment reference severe accident capability.” GEH clarified the reference to severe accident capability as those postulated for severe accident conditions as described in DCD Tier 2, Revision 4, Appendix 19B. For the postulated severe accident conditions, the service Level C pressure capacity for the PCCS heat exchangers at the temperature of 260 degrees C (500 degrees F) is 1.33 megapascals (MPa) gauge (193 psig). For comparison, the ESBWR containment design pressure is 0.312 MPa gauge (45 psig.)

The PCCS consists of six, low-pressure, separate loops sharing a common cooling pool. Each loop contains a two-module steam condenser (PCC condenser) designed to reject up to 7.8 megawatts thermal (MWt) of heat.

Following a postulated accident, after initial energy deposition into the pressure suppression pool, the PCCS keeps the containment pressure below its design limit for at least 72 hours, without water makeup to the IC/PCC pool, and beyond 72 hours with pool makeup.

The PCCS is open to the containment and receives a steam-gas mixture supply directly from the drywell. The condensed steam is drained to a GDCS pool, and the gas is vented through the vent line, which is submerged in the pressure suppression pool.

The PCCS operates in two distinct modes: a condensing mode and a pressure differential mode. Its operation is initiated by the difference in pressure between the drywell and the wetwell. Once a sufficient rate of steam condensation is established, the pressure inside the PCCS tubes is lower than the pressure in the drywell, which causes the flow of the steam-gas mixture into the heat exchange units. The condensate is then drained by gravity to a GDCS pool, and the noncondensable gases are collected in the lower drum of the PCCS units until its pressure exceeds the submergence head of the PCCS vent pipes in the suppression pool.

In the pressure differential mode, a pressure buildup in the drywell, caused by insufficient steam condensation inside the PCC condenser, will force flow through the PCCS, which pushes the noncondensable gases and the noncondensed steam into the suppression pool and potentially reestablishes the condensing mode of operation. This pressure buildup has to be greater than the submergence of PCCS vent pipes but not sufficient to clear the main vents. For that reason, the PCC vent line outlet is 0.85 m (2.8 ft) higher than the outlet of the upper horizontal main vents.

Since PCCS operation is completely passive, there is no need for sensing, control, logic, or power-actuated devices to function. GEH considers the PCCS condensers as an extension of the safety-related containment and thus not in need of isolation valves.

6.2.2.3 *Staff Evaluation*

The staff relied on the guidance in SRP Section 6.2.2, Revision 5, issued March 2007, to perform its review.

GDC 38 states, in part, "The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels." The TRACG results indicate that containment pressure is still rising at 72 hours, and the PCCS does not appear to rapidly reduce containment pressure and temperature as evident from the TRACG results presented in DCD Tier 2, Revision 3, Section 6.2. The applicant needed to demonstrate how the ESBWR meets the safety function of GDC 38. The staff issued RAI 6.2-139 to address this issue.

In response to RAI 6.2-139 (and RAI 6.2-140, discussed in Section 6.2.1.1.3 of this report), GEH made design modifications by adding a passive autocatalytic recombiner system (PARS) and PCCS vent fans, including power supplies. The following describes the staff's evaluation of these design modifications with respect to GDC 38.

The ESBWR pressure suppression concept employs a drywell that houses the nuclear system and a large volume of water outside the drywell called the suppression pool. If a LOCA occurs within the drywell, the pressure suppression system rapidly condenses the steam that is released through the break or that is generated by flashing of water which is released through the break to prevent overpressurization. Pressurization of the drywell results in venting of steam to the suppression pool where it is condensed, thus relieving pressure in the drywell. Decay heat in the core continues to generate steam, which is released into containment through the break. The PCCS removes heat from containment by condensing steam in the drywell. Condensate from the PCCS drains into the GDCS tanks, which provide water to the RPV for cooling the reactor core.

The rate of decay heat generation in excess of the rate of PCCS heat removal causes ESBWR containment pressurization. The design-basis analysis assumes both steam bypass of the suppression pool and radiolytic generation of noncondensable gases. Containment pressure, calculated using a conservative rate of decay heat generation, a bounding value of the steam bypass, and a conservative rate of radiolysis, continues to rise for 72 hours after a LOCA. However, containment pressure remains below the containment design pressure during this time. During the first 72 hours, the PCCS operates without need for active systems, electric power, or operator actions. The staff finds that the PCCS offers potential advantages over current active containment cooling systems and can provide sufficient containment heat removal to maintain containment pressure below its design value during this time.

Beyond 72 hours after a LOCA, the following additional systems supplement the PCCS to continue containment heat removal:

- Systems, structures, and components required for IC/PCC pool refill, including power supplies

- The PARS, which is conservatively assumed not to function until 72 hours, and then is assumed to function only to recombine hydrogen from radiolysis from 72 hours on (i.e., hydrogen content at 72 hours is assumed to remain constant for the duration of the LOCA recovery period)

- PCCS vent fans, including power supplies

Note that the PARS would remove hydrogen by initiating its chemical reaction with oxygen to produce steam, which can be condensed by the PCCS, helping to reduce the containment pressure. This reaction generates heat, countering the benefit of removing hydrogen and oxygen in the containment atmosphere. However, the net result is a drop in containment pressure because the PCCS can remove heat by condensing steam in the containment atmosphere. The PCCS and the additional systems can continue to remove heat from containment, maintaining its pressure below the design value up to 30 days and beyond. Two systems, (1) suppression pool cooling with a crosstie of the FAPCS and the RWCU/SDC heat exchanger and (2) the FAPCS in LPCI mode, will be available after 8 days following a LOCA, if needed to further reduce containment pressure.

In response to RAI 6.2-139, GEH stated the following:

> The analysis results indicate that the [drywell] pressure remains below the design pressure of 413.7 KPa (60 psia) for the first 72 hours after the [main steamline break accident], and then rapidly reduces and maintains the reduction with the refill of the [isolation condenser]/PCC pool and operation of the PCCS Vent Fans, achieving even lower pressures when the PARS were credited.

ESBWR containment pressure after a LOCA differs from that of operating BWR plants in all of the following ways:

- ESBWR pressure has a maximum value at 3 days, while operating BWR pressures peak within a few hours.

- The magnitude of ESBWR pressure drop at 3 days is lower than that for the operating BWRs.

- ESBWR pressure remains at elevated values in the long term compared to operating BWRs.

The staff concludes that the ESBWR does not reduce the containment pressure to as low a level as operating BWRs, but the ESBWR does provide adequate containment heat removal and meets the intent of GDC 38 because of the following:

- The PCCS can remove heat from containment and can maintain containment pressure below its design value without operator action or using active systems or electric power for 72 hours after a LOCA.

- The PCCS and additional systems can continue removing heat from containment from 3 days to beyond 30 days after a LOCA to maintain containment pressure below its design value.

- Systems are available after 8 days following a LOCA to further reduce containment pressure and to take the reactor coolant system to cold shutdown conditions, if needed.

The staff interpretation of GDC 38 applies specifically to the ESBWR passive design but, potentially, also to other similar passive safety systems.

The applicant's response, including design changes, addresses the staff's concern and is acceptable because the ESBWR does provide adequate containment heat removal and meets the intent of GDC 38. RAI 6.2-139 is therefore resolved.

The ESBWR PCCS is a safety-related ESF, which does not involve pumps, sprays, or fan coolers. Its design pressure is 758.5 kPaG (110 psig), compared to the containment design pressure of 310 kPaG (45 psig), and its design temperature is 171 degrees C (340 degrees F), the same as that for the containment. DCD Tier 2, Table 6.2-1 provides the containment design parameters.

Since PCCS operation is completely passive, there is no need for sensing, control, logic, or power-actuated devices to function. GEH considers the PCCS condensers as an extension of the safety-related containment drywell pressure boundary and thus not needing isolation valves. The staff evaluated the GEH position in Section 6.2.4.3 of this report under RAI 6.2-102 and finds it acceptable.

The PCCS operates in two distinct modes, a condensing mode and a pressure differential mode. In the pressure differential mode, a pressure buildup in the drywell, caused by insufficient steam condensation inside the PCC condenser, will force flow through the PCCS, which pushes the noncondensable gases and the noncondensed steam into the suppression pool and potentially reestablishes the condensing mode of operation. This pressure buildup has to be greater than the submergence of PCCS vent pipes but not sufficient to clear the main vents. For that reason, the PCC vent line outlet is 0.85 m (2.8 ft) higher than the outlet of the upper horizontal main vents. This is a critical elevation that should be verified by ITAAC and described in Tier 1 and Tier 2 of the DCD. DCD Tier 2, Revision 4, Section 6.2.2, did not include or describe the elevation of the PCC vent line relative to the upper horizontal main vents. Therefore, the staff issued RAI 6.2-169 to request this information. RAI 6.2-169 was being tracked as an open item in the SER with open items.

In response, GEH updated DCD Tier 2, Revision 5, Section 6.2.2.2.2, to state that "the vent line discharge point is set at an elevation submerged below low water level and at least 0.85 m

(33.5 in) and no greater than 0.900 m (35.4 in) above the top of the uppermost horizontal vent." GEH also added an ITAAC to verify the PCC vent line outlet elevation. These modifications address the staff's concerns. Based on the applicant's response, RAI 6.2-169 is resolved.

The PCCS is designed to seismic Category I, as described in RG 1.29 and ASME Code, Section III, Class 2, and Section XI requirements, to meet GDC 16 in Appendix A to 10 CFR Part 50. The material used must be a nuclear-grade stainless steel or equivalent material, which is not susceptible to intergranular SCC.

The six PCCS loops are each designed to remove 7.8 MWt of latent heat during condensation of pure steam inside the tubes at a pressure of 308 kPa absolute (45 psia) and a temperature of 134 degrees C (273.2 degrees F), with an outside pool water temperature of 102 degrees C (215.6 degrees F). For the steam-gas mixture and/or at the lower pressure and temperature, the condensing power of the condenser is lower. DCD Tier 2, Revision 9, Table 6.2-10 indicates the PCC design parameters.

To demonstrate PCCS performance at various flow rates, steam-gas compositions, and thermal conditions, a comprehensive testing program was developed to provide an experimental database for validation of analytical models. The staff reviewed and approved the PCCS-related test program in Chapter 21 of this report. The following briefly describes the three major tests (i.e., PANTHERS/PCC, PANDA, and GIRAFFE).

PANTHERS/PCC is an FS, two-module test facility at the Società Informazioni Esperienze Termoidrauliche (SIET) laboratory in Piacenza, Italy. Of the 63 tests performed using a prototypical heat exchanger, 13 were steady-state steam-only tests, 42 were air-steam tests, and 8 were noncondensable gas buildup tests with air, helium, and a mixture of both. The test matrix covered the range of expected accident conditions (pressure, temperature, and flow rates) as predicted by TRACG calculations. The tests confirmed the expected performance of the PCC condenser.

PANDA is a 1:25 scale (by volume), full-height integral systems test facility at the Paul Scherrer Institute in Switzerland. The PANDA test facility was configured to represent all major ESBWR containment components. It includes three full-height, scaled (by number of tubes) PCC condensers and one scaled IC unit. Of the 22 tests performed, 10 were steady-state, covering a wide range of expected steam flow and airflow rates, and 12 were transient tests, representative of various post-LOCA conditions. The tests confirmed the expected performance of the PCCS.

GIRAFFE is a full-height, small-scale (1:400 by volume) test facility at the Toshiba laboratories in Japan. The PCC condenser is represented by three full-height tubes. The main purpose of the tests was to demonstrate the effect of lighter-than-steam and heavier-than-steam noncondensable gases. Four tests were performed using nitrogen and helium. The tests confirmed that the PCCS can successfully operate in the presence of noncondensable gases.

The staff visited all of these facilities and performed several reviews of the engineering abilities of the personnel involved, testing equipment, and applied QA programs. The staff audited the QA programs and finds them acceptable, as discussed in Section 21.7 of this report. Therefore, the staff accepted the use of the test results to demonstrate PCCS performance and to support the verification and validation of the relevant analytical models.

The staff also performed its own independent studies of the PCCS performance at the Purdue University Multi-dimensional Integral Test Assembly (PUMA) facility. PUMA is a scaled (1:400 by volume, 1:4 reduced height) integral representation of the SBWR design similar to the PANDA facility. One of the purposes of these studies was to examine the effect of different scaling approaches. Unlike the PANDA facility, which preserves full height, the PUMA facility preserves the aspect ratio. This feature of PUMA provides additional insights into the multidimensional effects of an SBWR-like design. The PUMA tests qualitatively confirmed the PANDA results.

In the DCD, GEH did not describe the ESBWR test program as applied to the safety evaluation of the containment heat removal system. The staff requested this information in RAI 6.2-172. The GEH response was acceptable; however, the staff needed to verify that the response is incorporated in a future revision of the DCD. RAI 6.2-172 was being tracked as a confirmatory item. The staff confirmed that the description is included in the TRACG qualification report, NEDC-32725P, Revision 1, August 2002, “TRACG Qualification for SBWR,” which was reviewed and approved separately by the staff (see Conclusions 6.2.1.6.4). Therefore, RAI 6.2-172 is resolved.

In the DCD, GEH did not include an evaluation of GL 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” dated September 30, 1996, as indicated in DCD Appendix 1C, Table 1C-1. In RAI 6.2-170, the staff requested that GEH provide this discussion. In response, GEH explained that except for the containment isolation function, the chilled water system (CWS) equipment is all nonsafety-related and is not required to function during the response to a DBA. It is assumed that the nonsafety-related seismic Category II coolant boundary of the CWS or drywell cooling system heat exchanger may fail, opening to the containment atmosphere. Thus, the concerns of GL 96-06 have been considered in the design of the CWS and do not adversely affect the ESBWR response to a DBA.

During DBA conditions, the design feature providing cooling of the containment air for the ESBWR is the PCCS condensers, which condense steam that has been released to the drywell following a LOCA or MSLB to transfer the heat to the IC/PCC pools. The IC/PCC pools are designed to boil in order to perform their heat removal function. DCD Tier 2, Revision 3, Section 6.2.1, discusses the role of the PCCS condensers in maintaining containment pressure and temperature within design limits during DBAs and provides information about the function of the PCCS. DCD Tier 2, Revision 3, Section 6.2.2, gives the design details for the PCCS. The passive nature of the PCCS design prevents it from being subject to water hammer effects or thermally induced overpressurization.

Based on the GEH response to GL 96-06 and the passive nature of the PCCS design, the staff finds the GEH response acceptable; however, the staff needed to verify that the proposed revision to the DCD is incorporated in a future revision of the DCD. RAI 6.2-170 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD Tier 2, Revision 4, Table 1.1C-1. Based on the applicant’s response, RAI 6.2-170 is resolved.

DCD Tier 2, Revision 3, Section 1.11, Table 1.11-1, states that DCD Tier 2, Sections 6.2.2, 7.3.2, 9.2.7, and 9.4.8 address the evaluation of Task Action Plan Item B-12, “Containment Cooling Requirements (Non-LOCA).” The staff could not locate this discussion in Section 6.2.2 and requested, in RAI 6.2-171, that the applicant address Task Action Plan B-12.

In response, GEH stated that it referenced DCD Tier 2, Revision 3, Sections 6.2.2 and 7.3.2, because they describe the design of the PCCS, which performs the safety-related containment cooling for the ESBWR. In DCD Tier 2, Revision 3, Sections 9.2.7 and 9.4.8 have been referenced because they describe the design of the CWS and drywell cooling system (DCS), respectively. The CWS and DCS perform containment air cooling during normal operation and are isolated on a LOCA signal. A loss of normal containment cooling does not affect the operability of the safety-related PCCS to perform this function or the ability to place the ESBWR in a safe-shutdown condition. The PCCS is a passive system that does not have instrumentation, control logic, or power-actuated valves, and it does not need or use electrical power for its operation.

The staff finds the GEH response acceptable; however, it needed to verify that the proposed revision to the DCD is incorporated in a future revision of the DCD. RAI 6.2-171 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the changes were incorporated in DCD Tier 2, Revision 4, Table 1.11-1. Based on the applicant's response, RAI 6.2-171 is resolved.

In RAI 6.2-202, the staff requested that GEH address the possible accumulation of high concentrations of hydrogen and oxygen in the PCCS and ICS to meet 10 CFR 50.46(b)(5).

PCCS

During the blowdown period of a LOCA, most of the nitrogen in the drywell of the ESBWR would relocate into the wetwell airspace. Radiolysis in the core generates hydrogen and oxygen at the stoichiometric ratio, which would be released into the drywell with steam. A mixture of steam, hydrogen, oxygen, and any nitrogen remaining in the drywell would be drawn into the PCCS where steam is condensed, leaving mainly hydrogen and oxygen in the PCCS. Although a part of the hydrogen and oxygen that accumulates in the PCCS would relocate to the wetwell airspace through the PCC vent line, it is possible for the remaining hydrogen and oxygen to reach concentration levels that supports detonation.

In response, GEH agreed with the staff on the possibility of radiolytically generated hydrogen and oxygen accumulating in the PCCS at detonable levels following a LOCA and designed the PCCS to be able to perform its safety function after undergoing multiple hydrogen detonations. The GEH licensing topical report, NEDE-33572P, Revision 3, "ESBWR ICS and PCCS Condenser Combustible Gas Mitigation and Structural Evaluation," September 2010, describes PCCS design changes and the methodology by which the detonation loads were calculated. GEH modified the design of PCCS tubes, lower drum, and vent and drain lines. GEH did not evaluate the steam supply line and upper drums for detonation loading because the hydrogen and oxygen concentrations in those components would be low and would not support combustion as they are constantly being flushed by steam coming from the drywell. The staff finds that the GEH design of PCCS intake pipe and the upper drum is acceptable because the dilution by high steam concentration would prohibit detonation of hydrogen.

The following is a summary of PCCS design changes:

- Changed condenser tubing material from SA-213 Gr TP304L to SA-312 Gr XM-19 and increased the tube thickness to withstand detonation loading

- Increased the number of tubes per module (each PCCS condenser consists of two modules) to compensate for the reduction of heat transfer due to increased tube thickness and reduced thermal conductivity of the new material

- Increased thickness of the lower drum and changed the material to SA-182 Gr XM-19 to withstand detonation loading

- Added a safety-related catalyst module with platinum or palladium coated plates to the vent lines in the lower drum of the condenser to limit hydrogen and oxygen concentrations in the vent lines to below a detonable level

- Increased the thickness of the vent lines to withstand (1) pressure loading on the exterior of the vent line from a detonation occurring in a drain line and (2) high pressure generated by expansion of the post combustion gas mixture from a detonation postulated to occur in the lower drum

- Increased the thickness of the drain lines to withstand detonation loading

In calculating detonation loading for the PCCS tubes, lower drum, and drain line, GEH assumed a theoretical maximum concentration of hydrogen and oxygen at a stoichiometric ratio of 2:1. The staff finds that this treatment is conservative for mixtures in which the flame accelerates from deflagration to detonation (DDT) in a short period. However, with the introduction of inert gasses or vapors, the acceleration of the flame front may be delayed causing delayed DDT that can generate higher detonation pressures. During delayed DDT, the deflagration front undergoes a substantial acceleration period before transitioning to a detonation, or when the un-burnt mixture is compressed due to obstructions or closed ends in the structure. This compression at the onset of detonation has the potential to cause much higher localized pressure loads. To address the staff's concern, GEH noted that the detonation cell size for a hydrogen-oxygen mixture is too small to support delayed DDT. After reviewing the GEH response, the staff finds that delayed DDT would not be a concern for PCCS components. Therefore, the staff finds that the hydrogen and oxygen concentrations used by GEH to calculate detonation pressure loading are acceptable.

GEH calculated a bounding detonation pressure for a stoichiometric mixture of hydrogen and oxygen using the highest peak pressure that occurs during a loss of coolant accident (LOCA). GEH then applied the detonation pressure statically using dynamic load factors (DLF) in a finite element (FE) model for the PCCS condenser using the ANSYS computer code. GEH determined the resultant pressure following the passage of a detonation wave, which is called the Chapman-Jouguet (CJ) pressure, using a correlation between the CJ pressure and the initial pressure prior to detonation as given in a 2006 publication by J. E. Shepherd, "Structural Response of Piping to Internal Gas Detonation." The correlation is dependent on the composition of the fuel-oxidizer mixture, the initial conditions (pressure and temperature), and the geometry of the system. GEH used a CJ pressure ratio of 19. The staff determined that the GEH use of a stoichiometric mixture of hydrogen and oxygen, the peak LOCA pressure, and a temperature which is lower than that is expected in the PCCS during a LOCA would conservatively give high CJ pressures. Therefore, the staff finds that the GEH use of a CJ pressure ratio of 19 is acceptable.

The presence of bends, constrictions, and closed ends creates opportunities for reflections that can create localized peak pressures in excess of the CJ pressure. Based on a 1991 publication by J. E. Shepherd, et al., "Shock Waves Produced by Reflected Detonations," GEH assumed a peak pressure for a closed volume as a maximum of 2.5 times the CJ pressure. The staff finds that using a factor of 2.5 corresponding to a reflection by a closed end is conservative because bends and constrictions would generate lower pressure peaks.

GEH used a CJ pressure ratio of 19, as described above, combined with a DLF of 2. GEH determined DLF based on the 2006 publication by J. E. Shepherd. According to this publication, DLF of 2 can be used when the detonation velocity is not near the structural resonance velocity. Diluents, such as steam, cause the detonation velocity to drop, affording the possibility that the detonation velocity would come close to the resonance velocity of the component, in which case a DLF of 4 should be used. However, with addition of diluents the CJ pressure ratio also drops. GEH showed in Section 2.2.2.2 of NEDE-33572P that for a DLF of 4, the CJ value needs to be modified when the steam concentration is above 65 percent. At a steam concentration of 65 percent, a CJ pressure ratio would be 9.3. Thus, the product of CJ pressure ratio and DLF (i.e., 9.3 x 4) will be lower than that assumed in the design (i.e., 19 x 2). Based on the GEH determination of DLF and the staff's confirmatory calculations, the staff finds that a DLF of 2 is acceptable.

As described above, based on its review and confirmatory calculations, the staff finds that the GEH calculation of detonation pressure loading as used in the PCCS design is acceptable.

GEH proposed to revise the DCD, to include the following:

> To prevent the accumulation of combustible gas in the PCCS vent lines, catalyst modules containing metal parallel plates coated with platinum/palladium catalyst are placed at the entrance to the vent line, within each lower drum. These safety-related vent line catalyst modules are seismic category I and are environmentally qualified for the harsh post-accident environment in combination with the operating conditions of catalytic recombination, given their 60 year design life. The vent line catalyst modules are designed and built to withstand detonation loading in combination with other applicable dynamic loads, without losing their catalytic recombination functionality or negatively impacting the venting capability of the condenser.

After reviewing the proposed revision to the DCD and NEDE-33572P, the staff finds that the catalyst module added to the vent line in the lower drum of the condenser would limit hydrogen concentration in the vent line to below detonable level. Therefore, the staff finds that the GEH decision to ignore detonations in the PCCS vent line is acceptable.

With regard to PCCS performance, NEDE-33572P, Revision 1, states that the increase in PCCS tube thickness and change in the material will increase conduction resistance through the tube wall, which will have a negative effect on the overall heat transfer coefficient of the PCCS. To compensate for this effect, based on TRACG evaluations, GEH increased the number of tubes per PCCS module in order to keep the containment pressure response bounded by the values described in DCD Revision 7. To evaluate the effect of PCCS design changes on its heat transfer capability, in RAI 6.2-202 S01, the staff requested GEH to (1) confirm that TRACG validation for calculating PCCS heat transfer is applicable to the new design; (2) provide the results of TRACG analysis confirming that the containment pressure is bounded by values presented in DCD Revision 7; and (3) confirm that scaling groups used in ESBWR Scaling Report, Revision 2, NEDC-33082P, April 2008, are still applicable to the new design.

In response, GEH stated that (1) the PANTHERS and PANDA qualification tests as documented in NEDC 32725P, Revision 1, for TRACG validation are still applicable to the new PCCS design considering the different tube material, tube thickness, tube internal diameter, and different number of tubes; (2) the overall changes to the PCCS design have a relatively small impact on the overall heat transfer and the PCCS performance and; (3) GEH provided an evaluation showing the scaling groups remain applicable.

After reviewing the response, the staff finds that TRACG validation for calculating PCCS heat transfer is applicable to the new design.

In the above response, GEH provided the results of the TRACG analysis that includes the change in tube material, number of tubes, tube thickness, and tube inner diameter. Containment pressure and PCCS heat removal rate as predicted by TRACG for the MSLB bounding case as provided in DCD Revision 7 did not show any appreciable differences. Therefore, the staff finds that after PCCS design changes, the containment pressure is bounded by values presented in DCD Revision 7.

The results of the GEH calculations verify that the change in PCCS condenser response time due to the design differences is insignificant for the very slow, long-term containment pressure response, which is on the order of several hours (100,000 seconds), as discussed in NEDC-33082P. The results further demonstrate that the overall changes to the PCCS design have a relatively small impact on the overall heat transfer. Calculations show that the differences between overall heat transfer coefficient, total thermal resistance, fluid transport time and thermal time constant of the tube wall for the two designs are not significant. From a scaling perspective, these changes are within the same order of magnitude (i.e., within the acceptable range) as those for the ESBWR test program, which is discussed in Chapter 21 of this report.

Thus, the local or “bottom-up” scaling shows that PANTHERS tests for PCCS are still applicable to the new design since the PCCS overall heat transfer has not changed. Therefore, the Pi-groups for the “top-down” scaling groups are expected to remain the same, and no change is necessary to the scaling groups. As a result, the analysis confirms that the modified PCCS design satisfies the scaling criterion that was used for the ESBWR test program. In addition, the staff believes that the changes in the PCCS design are not expected to create any new or different phenomena that were not observed in the test.

On the basis of the discussion made above, the staff finds the GEH response acceptable. The staff, therefore, concludes that there is reasonable assurance that the PANTHERS/PCCS test data continue to be relevant and sufficient to apply TRACG for the modified PCCS Condenser design.

Structural Analysis of PCCS

The PCCS condensers were designed as part of the containment pressure boundary according to ASME Code, Section III, Subsection NE. Therefore, under Section 3.8.2 of this report, the staff evaluated the structural integrity of the PCCS within the jurisdictional boundary of ASME Code, Subsection NE; in particular, the staff evaluated the capability of the PCCS to withstand the effects of deflagration or detonation of non-condensable gases during the 72 hour-period associated with a LOCA.

On September 22, 2010, staff conducted an audit of supporting calculations and the basis for the GEH licensing topical report NEDE-33572P at the Nuclear Energy Institute (NEI) office in

Rockville, Maryland. During this audit, the NRC team reviewed calculations associated with the structural analysis of the PCCS to withstand detonation loads, to obtain reasonable assurance that the design is in conformance with the ASME Code, Subsection NE, and the guidance in SRP 3.8.2 – See "Summary of Audit for Review of License Topical Report NEDE-33572P, Revision 2, Appendix C and Supporting Analyses," September 22, 2010.

To resolve the remaining issues, the applicant responded to RAI 6.2-202 S01, by providing details of its structural evaluation in Appendix B and Appendix C of NEDE-33572P. The information included in these Appendices addresses the staff's concerns as described below:

The applicant determined, and the staff agrees, that the appropriate acceptance criterion to be used in the PCCS structural design, for load combinations including detonation loads, was Service Level C per the ASME Code, Section III, Subsection NE. It was not clear to the staff if all PCCS components within the jurisdictional boundary of ASME Code, Subsection NE, were designed to this criterion. The staff requested that the applicant confirm that all PCCS components within the containment boundary were designed using acceptance criteria for Service Level C. In response, the applicant confirmed that the design of each critical PCCS component within the jurisdictional boundary of ASME Code, Subsection NE, was modified to satisfy the corresponding allowable stress limits for Service Level C. Therefore, this item is resolved.

The structural analysis of critical PCCS components, for detonation loads, followed an equivalent-static approach in which detonation pressures were statically applied to FE sub-models. All dynamic effects, including the effects of pressure wave reflections, were accounted for by using appropriate amplification factors. However, this equivalent-static approach did not address the dynamic effects of detonation loads on the entire PCCS assembly, including its supporting structure and anchorage. In RAI 6.2-202 S01, the staff requested that the applicant assess and include in its analysis and design the effects of detonations on the entire PCCS assembly, including its support structure and anchorage.

In response, the applicant performed an additional dynamic FE analysis to evaluate the effects of detonation loads on the entire PCCS assembly, including its supporting structure and anchorage. The dynamic analysis was performed by applying a spatially varying pressure time-history to the interior of the lower drum. This time-history represents the effect of a one-dimensional detonation pressure wave front initiating at one end of the lower drum, propagating along the length of the lower drum, and eventually reaching an internal equilibrium state. An appropriate factor was considered to account for reflections of the pressure wave-front inside the lower drum. The applicant included the analysis method and results in Appendix B and Appendix C of the LTR. The staff reviewed the analysis method and the results presented by the applicant and considered them acceptable. The analysis appropriately considered the dynamic effects of detonations on the entire PCCS assembly by applying the dynamic pressure loads to the most critical area of the PCCS and evaluating its effects by a FE time-history analysis. For the design of the various components and supports of the PCCS, the applicant also appropriately considered the stresses and reaction loads from the aforementioned analysis. Therefore this item is resolved.

It was not clear to the staff that thermal effects following a detonation were accounted for in the structural design – particularly the thermal effects on the condenser tubes, which are slender elements restrained against longitudinal expansion. In response to RAI 6.2-202 S01, the applicant performed additional calculations to demonstrate that post-detonation thermal stresses induced in the condenser tubes are bounded by stresses due to detonation loads. The applicant added Section 2.2.7 to the LTR to document the results of this evaluation. Since the stresses due to the post-detonation thermal effects are bounded by the stresses due to detonation loads, this item is resolved.

Since the number of detonations expected to occur during the 72 hour-period associated with a LOCA could be high, the applicant was also asked in RAI 6.2-202 S01, to perform a fatigue evaluation for the total number of expected stress cycles. In response, the applicant performed a simplified fatigue evaluation of all critical PCCS components. The applicant demonstrated, and the staff agrees that the corresponding usage factors were sufficiently lower than 1.0 in all cases. Therefore, this item is resolved.

ICS

Similar to hydrogen accumulation in PCCS, there is a potential for hydrogen accumulation in the ICS tubes during post-LOCA conditions. In LTR NEDE-33572P, Section 4.2, GEH stated that during a LOCA event, the ICS injection is credited using the condensate stored in its drain piping. The heat removal through the ICS condenser is not credited for LOCA. However, there is potential for condensation to occur, and given enough time it is possible for combustible gases to accumulate in the ICS condenser to a detonable level following a LOCA. In order to prevent this buildup from occurring, a logic change was implemented for the ICS steam admission isolation valves in which the valves now automatically close after receiving an indication that the DPV have opened. The staff agrees with the applicant that closing the ICS steam admission isolation valves when the RPV is depressurized mitigates the accumulation of hydrogen and oxygen.

The applicant states that a TRACG evaluation shows that once it is isolated from the vessel, the ICS condenser pressure will drop below 0.1 MPa absolute (15 psia) from the reactor operating pressure within 2,000 seconds, and noncondensable gas partial pressure will not exceed 0.63 MPa (91 psia) following isolation. The applicant also stated that detonation under these conditions is highly unlikely; however, if one were to occur, the resulting loads would be within the original design pressure 8.62 MPaG (1250 psig) of the ICS. The methodology by which the PCCS CJ pressures were calculated is also applied to the ICS; however, credit is taken for the detonation properties of the mixture, which contains no less than 37 percent steam (based on the TRACG evaluation). As a result, GEH used a CJ pressure ratio of 13.3 corresponding to 20 percent steam present in the noncondensable gas mixture per Table 4-1 of NEDE-33572P and calculated the maximum detonation pressure to be 8.32 MPa absolute (1207 psia) at 75 seconds after isolation, which is below 8.62 MPa absolute (1250 psia). In addition, a fatigue evaluation will be conducted as part of the detailed design of the ICS and will be addressed in the design report for this component, as stated in ESBWR DCD Tier 1, Table 2.4.1-3, Design Commitment 2a1, in accordance with ASME Code Section III Division 1, Subsection NC "Class 2 Components Rules for Construction of Nuclear Facility Components." This is acceptable to the staff because the loads from a potential detonation do not exceed the original design pressure of the ICS.

For non-LOCA events such as station blackout (SBO), GEH proposed modifications to the condenser vent function in order to keep the unit continuously purged of noncondensable gas. A logic change was implemented in which the lower head vent valves automatically open six hours after the ICS is initiated regardless of the system pressure. Once open, the vent will bleed steam and noncondensables from the condenser to the suppression pool, keeping the steam fraction at high levels (beyond the detonation range) throughout the event. Also, the vent valves are designed to fail open on a loss of power to provide additional reliability for this function. In addition, a flow restrictor is included in the vent line to keep the condenser purged and maintain the RPV water above Level 1 for 72 hours.

In RAI 6.2-202 S01, the staff requested justification that the six-hour time delay would be short enough to preclude the accumulation of a detonable concentration of hydrogen and oxygen in the ICS. In response, GEH revised LTR NEDE-33572P Section 4, ICS Methodology, to provide the technical basis for the six-hour delay, and stated that ESBWR radiolytic hydrogen production calculation is consistent with the methodology of Appendix A to SRP Section 6.2.5 and RG 1.7, "Control of Combustible Gas Concentrations in Containment,". The staff reviewed the radiolytic gas production calculation results summary included in Section 4.1.2 of NEDE-33572P, Revision 3. The calculation results show very low gas production at six hours after SBO and consequently, hydrogen and oxygen concentrations are expected to be below the deflagration limits and hence acceptable, and therefore, the issue is resolved.

Based on the above evaluation the staff finds that the applicant has addressed the possible accumulation of high concentrations of hydrogen and oxygen in the PCCS and ICS. The applicant has used an acceptable methodology to calculate concentrations of hydrogen and oxygen, to calculate loads and load combinations, to calculate stresses which meet applicable ASME code requirements. Based on the above, RAI 6.2-202 is resolved.

6.2.2.4 *Conclusions*

The review of the ESBWR test program revealed that it correctly established the expected containment thermal conditions and the ranges of relevant parameters included in the experimental matrices. The test data appear to be of good engineering quality and sufficient to provide a basis for validation of TRACG analytical models, as well as for verification of the code predictions of containment behavior under various accident conditions. The staff accepts the TRACG prediction that, within 72 hours of the DBA, the ESBWR pressure and temperature during the postulated DBA scenarios are sufficiently within the design values.

6.2.3 Reactor Building Functional Design

The RB structure encloses all penetrations through the containment (except for those of the main steam tunnel and IC/PCC pools). The RB provides an added barrier to fission product released from the containment in case of an accident; contains, dilutes, and holds up any leakage from the containment; and houses safety-related systems.

6.2.3.1 *Regulatory Criteria*

The staff reviewed the RB in accordance with SRP Section 6.2.3 for secondary containment. The staff realized that the ESBWR design has significant differences from the secondary containment of currently operating BWR facilities. The staff evaluation discusses these differences. Conformance with these regulatory criteria forms the basis for determining the acceptability of the RB functional design. The staff also reviewed the subcompartment analyses in accordance with SRP Section 6.2.1.2, Revision 3, for containment integrity. The staff also reviewed the design with respect to the associated regulatory guidance and criteria.

- GDC 4, as it relates to safety-related SSCs being designed to accommodate the effects of normal operation, maintenance, testing and postulated accidents, and being protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures

- GDC 16, as it relates to reactor containment and associated systems being provided to establish essentially leaktight barriers against the uncontrolled release of radioactive material to the environment

- GDC 43, "Testing of Containment Atmosphere Cleanup Systems," as it relates to atmosphere cleanup systems having the design capability to permit periodic functional testing to ensure system integrity, the operability of active components, and the operability of the system as a whole and the performance of the operational sequence that brings the system into operation

- GDC 50, as it relates to the design of the containment internal compartments to ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident

- 10 CFR Part 50, Appendix J, as it relates to the secondary containment being designed to permit preoperational and periodic leakage rate testing so that bypass leakage paths are identified

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," as it relates to guidance in assumptions concerning mixing in the RB in applying the alternative source term

- SRP Section 6.2.3, as it provides methods acceptable to the staff for the review of secondary containments

- SRP Section 6.2.1.2, as it provides methods acceptable to the staff for the review of subcompartment analysis

- NUREG–1242, with specific references to passive plant designs

6.2.3.2 *Summary of Technical Information*

The RB structure encloses penetrations through the containment (except for those of the main steam tunnel and IC/PCC pools). The RB has the following functions:

- Provides an added barrier to fission product released from the containment in case of an accident

- Contains, dilutes, and holds up any leakage from the containment

- Houses safety-related systems.

The RB consists of rooms and compartments, which are served by one of the three ventilation subsystems: the contaminated area ventilation subsystem (CONAVS), refuel and pool area ventilation subsystem (REPAVS), and clean area ventilation subsystem (CLAVS). None of these compartmentalized areas communicates with any other.

Under accident conditions, the CONAVS and REPAVS areas of the RB automatically isolate on high radiation to provide a holdup volume for fission products. When isolated, the RB (CONAVS and REPAVS areas) can be serviced by the RB heating, ventilation, and air conditioning (HVAC) purge exhaust filter units. No credit is taken for the filters in dose consequence analyses. With low leakage and stagnant conditions, the basic mitigating function is the holdup of fission products in the RB CONAVS area itself. The ESBWR design does not include a secondary containment; however, in radiological analyses, credit is taken for the existence of the RB CONAVS area surrounding the primary containment vessel. RB CONAVS areas envelop all containment penetrations except penetration for main steam and feedwater lines located in the main steam tunnel. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.35 percent per day and RB CONAVS area leakage rate of 141.6 liters per second (l/s) (300 cubic feet per minute [cfm]), show that offsite and control room doses after an accident are less than allowable limits, as discussed in Chapter 15.

During normal plant operation, potentially contaminated areas within the RB are kept at a negative pressure with respect to the environment, while clean areas are maintained at positive pressure. The ESBWR does not need and thus has no filter system that performs a safety-related function following a DBA. Therefore, GEH indicated that GDC 43 is not applicable.

RB leakage less than the maximum leak rate used in the accident dose calculations has the potential to increase the radiation dose inside the RB following a DBA. The environmental qualification program addresses the evaluation of the effect of increased radiation levels on equipment, and the emergency planning program, through emergency operating procedures, addresses any increased hazards during postaccident RB reentry.

Personnel and equipment entrances to the RB consist of vestibules with interlocked doors and hatches. Large equipment access is by means of a dedicated, external access tower that provides the necessary interlocks.

Design Bases

The RB is designed to meet the following safety design bases:

- The RB maintains its integrity during the environmental conditions postulated for a DBA.
- The RB HVAC system (RBVS) subsystems (CONAVS and REPAVS) automatically isolate upon detection of high radiation levels in their respective ventilation exhaust system.
- Openings through the RB boundary, such as personnel and equipment doors, are closed during normal operation and after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms that are monitored in the control room.
- Detection and isolation capability for high-energy pipe breaks within the RB is provided.
- The compartments within the RB are designed to withstand the maximum pressure due to a high-energy line break (HELB). Each line break analyzed is a double-ended break. This analysis considers the rupture producing the greatest blowdown of mass and enthalpy in conjunction with the worst-case, single, active component failure. Blowout panels between compartments provide flow paths to relieve pressure

- The RB design allows for periodic testing to ensure that the leakage rates assumed in the radiological analyses are met. The radiological analyses assume that areas served by the RB CONAVS form this boundary.

Design Description

The RB is a reinforced concrete structure that forms an envelope completely surrounding the containment (except the basemat). During normal operation, the potentially contaminated areas in the RB are maintained at a slightly negative pressure relative to adjoining areas by the CONAVS portion of the RBVS. This ensures that any leakage from these areas is collected and treated before release. Airflow is from clean to potentially contaminated areas. Stack radiation monitors check RB effluents for radioactivity. If the radioactivity level rises above set levels, the discharge can be routed through the RB HVAC online purge exhaust filter unit system for treatment before further release.

Penetrations through the RB envelope are designed to minimize leakage. All piping and electrical penetrations are sealed for leakage. The RBVS is designed with safety-related isolation dampers and tested for isolation under various accident conditions.

HELBs in any of the RB compartments do not require the building to be isolated. These breaks are detected and the broken pipe is isolated by the closure of system isolation valves. No significant release of radioactivity is postulated from these types of accidents because reactor fuel is not damaged.

The following paragraphs briefly describe the major compartments in the ESBWR design.

RWCU Equipment and Valve Rooms

The two independent RWCU divisions are located in the RB. The RWCU piping originates at the RPV. High-energy piping leads to the RWCU divisions through a dedicated, enclosed pipe chase. The steam/air mixture resulting from an HELB in any RWCU compartment is directed through adjoining compartments and the pipe chase to the RB operating floor. The design-basis break for the RWCU system compartment network is a double-ended break. The applicant provided pressure profiles for all postulated RWCU/SDC system break cases for each individual room or region. The envelope profile represents the calculated maximum pressure response values for the given room or region due to all postulated RWCU/SDC system pipe breaks. These pressure profiles include no margin.

Isolation Condenser System

The ICs are located in the RB. The IC steam supply line is connected directly to the RPV. The supply line leads to a steam distribution header, which feeds four pipes. Each pipe has a flow limiter to mitigate the consequences of an IC line break. The IC design-basis break is a double-ended break in the piping after the steam header and flow restrictors. The IC/PCC pool is vented to atmosphere to remove steam generated in the IC pools by the condenser operation. In the event of an IC break, the steam/air mixture is expected to preferentially exhaust through hatches in the refueling floor and into the RB operating area with portions of the steam directed through the pool compartments to the stack, which is vented to the atmosphere. Because the vent path through the hatches leads to the refueling floor area, which is a large open space with no safety implications, the pressurization analysis excluded this event.

Main Steam Tunnel

The RB main steam tunnel is located between the primary containment vessel and the turbine building (TB). The limiting break is an MSL longitudinal break. The MSLs originate at the RPV and are routed through the steam tunnel to the TB. The steam/air mixture resulting from an MSLB is directed to the TB through the steam tunnel.

No blowout panels are required in the steam tunnel because the flow path between the steam tunnel and the TB is open.

Design Evaluation

Fission Product Containment

Sufficient water is stored within the containment to cover the core during both the blowdown phase of a LOCA and during the long-term post-blowdown condition. Because of this continuous core cooling, fuel damage resulting in fission product release is a very low probability event. If there is a release from the fuel, most fission products are readily trapped in water. Consequently, the large volume of water in the containment is expected to be an effective fission product scrubbing and retention mechanism. Also, because the containment is located entirely within the RB, multiple structural barriers exist between the containment and the environment. Therefore, fission product leakage from the RB is mitigated.

Compartment Pressurization Analysis

RWCU pipe breaks in the RB and outside the containment were postulated and analyzed at 102 percent power and 187.8 degrees C (370 degrees F) feedwater temperature. For compartment pressurization analyses, HELB accidents are postulated as the result of piping failures in the RWCU system, where locations and size of breaks result in maximum pressure values. Calculated pressure responses have been considered in order to define the peak pressure of the RB compartments for structural design purposes. The calculated peak compartment pressures include a 10-percent margin. The maximum is 35.2 kPaG (5.11 psig), which is below the RB compartment pressurization design requirement. Values of the mass and energy releases produced by each break are in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS)-56.4, “Pressure and Temperature Transient Analysis for Light Water Reactor Containments.” The mass and energy blowdown from the postulated broken pipe terminates when system isolation valves are fully closed after receiving the pertinent isolation closure signal.

A conservative RWCU model based on RELAP5/Mod3.3 has been developed to evaluate the mass and energy release for five break locations. Total blowdown duration is based on the assumption that the isolation valve starts to close at 46 seconds (1 second instrument time plus 45 seconds built-in time delay in blowdown differential flow detection logic) after the break and the isolation valve is fully closed in 15 seconds.

After the initial inventory depletion period, the steady RPV blowdown is choked at the venturi located upstream of the isolation valve since the venturi flow area is smaller than the isolation valve flow area. After the isolation valve starts closing, as soon as the valve area becomes equal to the venturi flow area, the break flow is choked at the isolation valve. The break flow stops when the isolation valve is fully closed.

The narrative of the event described above applies to all five cases analyzed since the breaks are all located downstream of the isolation valve and the dynamics of the break responses are similar.

Subcompartment pressurization effects resulting from the postulated breaks of high-energy piping have been analyzed according to ANSI/ANS-56.10, "Subcompartment Pressure and Temperature Transient Analysis in LWRs." To calculate the pressure response in the RB and outside the containment resulting from HELB accidents, the analysis used the CONTAIN 2.0 code. The nodalization contains the rooms where breaks occur, and all interconnected rooms or regions through flow paths such as doors and hatches. The selected nodalization maximizes differential pressure. Owing to the geometry of the regions, each room or region was assigned to a node of the model. No simple or artificial divisions of rooms were considered to evaluate the sensitivity of the model to nodalization. A sensitivity study of pressure response was performed to select the time step. Additional sensitivity studies were performed to evaluate the impact of the heat sinks, dropout, and inertia term. Modeling follows the recommendations given by SMSAB-02-04, "CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations."

Tests and Inspections

Position status indication and alarms for doors, which are part of the RB envelope, are tested periodically. Leakage testing and inspection of other architectural openings are also performed regularly.

The RB (CONAVS area) can be periodically tested to ensure that the leakage rates assumed in the radiological analysis are met, as required by TS 3.6.3.1. RB exfiltration testing is a positive pressure test of the CONAVS volume to confirm that the test leak criteria bound the analytical limit derived in the dose modeling. A nominal ¼-inch water gauge (w.g.) differential pressure bounds the effects of worst-case wind loading applied across a face of the RB. Many pressure measurements are taken at designated areas, and interconnecting doors and dampers are opened to ensure that uniform pressure is established within the contaminated areas of the RB (CONAVS area). The RB exfiltration test pressure is maintained for sufficient time to ensure that steady-state conditions are established (approximately ½ hour to 1 hour). These RB exfiltration test leak rate acceptance criteria are adjusted based on the actual CONAVS area test differential pressure applied to ensure minimal impact of test parameter uncertainties (flow instrument uncertainty, CONAVS area temperature, and pressure gradients).

Instrumentation Requirements

DCD Tier 2, Revision 9, Section 7.3.3 gives details of the initiating signals for isolation. Doors that form part of the RB boundary are fitted with position status indication and alarms.

6.2.3.3 *Staff Evaluation*

The staff review focused on compliance with the GDC listed in Section 6.2.3.1.

GDC 4 states that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a LOCA. The staff issued RAI 6.2-155 to obtain information on how the ESBWR complies with GDC 4. RAI 6.2-155 was being tracked as an open item. In response, the applicant included in the DCD a description of analyses, such as pressurization due to high-pressure line break, and identified and stated that ITAAC in DCD Tier 1, Table 2.16.5-2, will verify compliance with GDC 4. The staff concluded that the design complies with the requirements of GDC 4 in that the applicant has shown by analysis that the plant is designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including a LOCA. Based on the applicant's response, which included information linked to DCD changes and ITAAC, the staff finds that this open item is resolved.

GDC 16 states that reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. In the ESBWR, the RB CONAVS serves as the barrier against uncontrolled release of radioactivity to the environment from primary containment leakage through penetrations. In accordance with the staff position stated in NUREG–1242, the RB CONAVS is considered to be a safety envelope that is a concrete and reinforced steel structure (secondary containment) within the RB that forms an envelope completely surrounding the primary containment. NUREG–1242 allows appropriate credit for fission product holdup without requiring that a negative pressure be maintained in the secondary containment if the secondary containment leakage and mixing performance are consistent with the values used by the staff in its radiological assessment.

The applicant stated that the ESBWR does not include a secondary containment; however, the applicant takes credit for the existence of the RB CONAVS area surrounding the primary containment vessel in radiological analyses. The staff finds that the RB CONAVS functions as the secondary containment by providing tight controls on leakage through concrete and steel construction, a periodic leakage test program, and holdup volumes, as the principal means of controlling radioactive release.

The staff considered the applicant's statement with respect to the applicability of GDC 16, particularly with respect to the control of leakage from the RB CONAVS to the environment, because of its significant impact on the design-basis analysis dose results. The staff's method for calculating dose results is the RADTRAD software that models releases from the facility and determines an integrated dose over 30 days at control room, exclusion area boundary, and low-population zone receptors. Compliance with GDC 16 requires the applicant to show that the secondary containment leakage and mixing performance are consistent with the values used by the staff in its radiological assessment. The secondary containment leakage is the exfiltration rate. The mixing performance is the percent of the secondary containment volume credited for dilution in the RADTRAD design-basis analysis.

The applicant established two parameters based on the RB CONAVS design that are used as direct inputs to the design-basis analysis: an exfiltration rate from the RB CONAVS to the environment of 141.6 l/s (300 cfm), and an effective mixing volume that is 50 percent of the RB CONAVS volume, which is used to determine the dilution of the source term that is being released. The applicant also stated that the source term entering the RB CONAVS would be 0.35-percent mass of the primary containment per day. DCD Tier 2, Revision 9, Table 15.4-5, documents these three parameters.

The applicant's basis for 141.6 l/s (300 cfm) exfiltration is a pressure test of the RB CONAVS volume, in which makeup airflow from a fan pressurizing the RB CONAVS is measured to be less than or equal to 141.6 l/s (300 cfm) as the RB CONAVS area is raised and maintained at ¼-inch w.g. positive pressure. In NUREG–1242, the staff agreed to consider holdup as a means to reduce releases to the environment, on the condition that the exfiltration rate be limited to 25-percent volume per day of the safety envelope volume. The RB CONAVS volume is the safety envelope volume. An exfiltration flow of 141.6 l/s (300 cfm) represents approximately 50-percent volume per day. Thus, the applicant is deviating from the staff position stated in NUREG–1242. The applicant's basis for the deviation is that it would be very difficult to conduct an accurate pressure test of a volume the size of the RB CONAVS with a maximum criterion of 25-percent volume per day.

The staff reviewed the deviation and acknowledges that it would be a difficult test situation. The staff is concerned that the quantity of holdup has not been explicitly established and would have a high degree of uncertainty. Keeping the exfiltration rate small lessens the impact of RB CONAVS releases to the environment due to the uncertainty in the holdup. The staff agreed to consider the increase in exfiltration rate, provided that the requirements of the design-basis dose analysis are met and the uncertainty in holdup is appropriately addressed.

The applicant's basis for an effective mixing volume of 50 percent of RB CONAVS volume is twofold: (1) a reference to RG 1.183 in which 50-percent mixing is permitted if adequate means can be shown to cause the environment to mix, and (2) a GOTHIC analysis which demonstrates that the actual release that occurs considering holdup is less than the release that results in the design-basis RADTRAD analysis using the 50-percent mixing volume, thus showing that the RADTRAD analysis is conservative.

The staff reviewed the reference to RG 1.183 and determined that it provides no justification for a 50-percent mixing rate for a passive design. RG 1.183 (Appendix A, paragraph 4.4) states that "credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise the leakage from the primary containment should be assumed to be transported directly to the exhaust systems without mixing." RG 1.183 clearly requires a means of mixing normally provided by the standby gas treatment system to take credit for 50-percent mixing. The applicant states in the DCD that RB CONAVS has low leakage and stagnant conditions, the exact opposite of a well-mixed environment.

The staff reviewed the arrangement and operation of the RB with respect to holdup and determined that the potential leakage from penetrations took place in penetration rooms that were concrete structures and were maintained closed by administrative controls and door alarms in the control room. Thus, if leakage occurred, it would build significantly in these penetration rooms before entering other parts of the RB CONAVS safety envelope. Before leakage from the RB CONAVS safety envelope would occur, levels of primary leakage would have to concentrate in order to be a significant contributor to the dose consequence analysis. The holdup time resulting from passing through multiple barriers provides for some decay of short-lived isotopes and ensures that a degree of mixing does in fact occur before release from the RB CONAVS safety envelope. Based on the robust concrete structures, closed penetrations rooms under administrative controls, and multiple barriers to release, the staff concludes that a 50-percent mixing assumption in the dose consequence analysis is reasonable.

The staff issued RAI 6.2-165 to obtain information on how the applicant established the assumption on mixing in the RB which is used in the DCD Tier 2, Chapter 15 dose consequence analysis. In the interim, design changes occurred that changed the safety envelope from the entire RB to the contamination portion only, the mixing assumption to 50 percent per day of the contaminated volume, the primary containment leakage to 0.35 percent per day of the containment volume and added administrative controls on contaminated area doors and other related changes documented in DCD Revisions 5 and 6. In response to RAI 6.2-165, the applicant submitted a GOTHIC analysis of the RB CONAVS volume to demonstrate that the releases from the RB CONAVS were significantly less than those determined by the RADTRAD dose consequence analysis using the 141.6 l/s (300-cfm) exfiltration and 50-percent mixing assumptions. The response included sensitivity studies and addressed uncertainties. The result of the analytical studies added credence to the determination that the 50-percent mixing assumption is acceptable.

RAI 6.2-165 was being tracked as an open item in the SER for open items. Based on the applicant's response, which provided information and insight into the holdup capabilities, and in consideration of other staff confirmatory evaluations, this open item is resolved.

Although GOTHIC is a powerful tool for analyzing conditions throughout a building, many parameters require assumptions or careful measurements to obtain the results and would require revalidation over time. The applicant also adjusted some of the parameters, such as door gaps and leakage points, and showed that the sensitivity of most of the parameters had only a small effect. The staff has not previously accepted the use of GOTHIC as an analysis tool for this application. The application of GOTHIC to this safety evaluation is accepted as collaborating information.

The staff accepted the 50-percent mixing volume for use in RADTRAD on the following bases:

- The staff's determination that significant holdup would occur because of the robust concrete building room structures that form multiple barriers to release to the environment.

- A test program that ensures that the RB CONAVS safety envelope leakage would not exceed the 141.6 l/s (300 cfm) criterion that is part of the dose consequence analysis assumptions.

- The 50-percent mixing volume for the RADTRAD analysis adds substantial conservatism and accounts for holdup distribution changes in the RB CONAVS as the result of infiltration/exfiltration flow.

- Analytical evaluations and sensitivity studies provided by the applicant are consistent with the staff's evaluation and indicate that changes in temperature, resistance factors, and penetration leakage points have minimal impact on results.

- Appropriate ITAAC and administrative controls have been established to ensure that the RB CONAVS is constructed and maintained in accordance with the evaluated design.

The staff finds that the applicant has complied with GDC 16 by providing in the design the means to prevent uncontrolled release to the environment of radioactive effluents through holdup and limited leakage. As such, the applicant has ensured that the guidance values and limits of the radiological consequence analyses are not exceeded.

SRP Section 6.2.3 references GDC 43 as applying to secondary containments and states that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to ensure: (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

The DCD states that during normal plant operation, potentially contaminated areas within the RB are kept at a negative pressure with respect to the environment, while clean areas are maintained at positive pressure. The ESBWR does not need, and thus does not have, a filter system that performs a safety-related function following a DBA. Therefore, the design criterion of GDC 43 is not applicable.

The staff issued RAI 6.2-166 to obtain information on how buildup of postaccident radiation in the RB is controlled and how it impacts access. In response, the applicant acknowledged that the absence of a standby gas treatment system allowed radiation levels to build in the contaminated portion of the RB after an accident and that these radiation levels could preclude entry for the purpose of making a cross-tie between the RWCU/SDC and the FAPCS to facilitate achieving cold shutdown. The applicant added a 472 l/s (1,000 cfm) RTNSS E filter system that could be used to clean up the contaminated portion of the RB after 72 hours. This system, the RB HVAC accident exhaust filter system, exhausts to the environment through the RB vent. The applicant evaluated the impact on the dose consequence analysis and determined that the results of the dose consequence analyses presented in DCD Tier 2, Chapter 15 bound the results of operation of this system on a parametric basis for all times greater than 8 hours into the accident. The applicant assigned a charcoal adsorber efficiency of 95 percent, based on compliance with 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, June 2001. The staff determined that if this system were to operate in the 30-day accident recovery period, it would impact the dose analysis which is safety-related, that it is acceptable for the system to be classified as RTNSS since its operation is not required in the timeframe of 0–72 hours, but that the filter testing should be done in accordance with 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, June 2001, since it provides filter efficiency parameters to the dose consequence analysis.

The applicant responded that the system is not required after the accident and that it provides defense-in-depth. Emergency operating procedures would control the operation of the system. These procedures would confirm that there is no adverse impact on the dose consequence analyses before their operation. In addition, the filters would be tested to the same test requirements specified in RG 1.52, but the system would retain its classification as a nonsafety system designed in accordance with RG 1.140. The staff concludes that the system facilitates the cleanup of the contaminated portion of the RB, does not impose any additional impact on release of radiation to the environment, and meets the requirements of GDC 43.

RAI 6.2-166 was being tracked as an open item. Based on the applicant’s response, which included design changes to add an RTNSS qualified filter system that could be used after an accident and providing additional assurance that dose levels defined in the radiological consequences analyses documented in Chapter 15 would not be exceeded, the staff finds that this open item is resolved.

GDC 50 states that the containment internal compartments will be designed to ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The staff issued RAI 6.2-46 and RAI 6.2-154, to obtain additional information for the purpose of conducting confirmatory evaluations. In response to these RAIs, the applicant presented analyses using NRC-approved codes to demonstrate that the containment internal compartments are designed to meet GDC 50. The staff conducted confirmatory calculations for HELBs caused by pipe failures in the RWCU system, which show that the applicant's peak pressure is conservative and is below the design value for peak pressurization. In the applicant's conservatively chosen HELB cases, the maximum pressure observed in internal compartments is 35.2 kPaG (5.1 psig), which is less than the applicant's design limit of 36 kPaG (5.2 psig).

RAI 6.2-46 and RAI 6.2-154 were being tracked as open items in the SER with open items. Based on the applicant's response, which included analyses using NRC-approved codes, the staff evaluated internal compartment pressures and temperatures and finds that these open items are resolved.

In Appendix J to 10 CFR Part 50, Option A states in Section IV.B that other structures of multiple barrier or sub-atmospheric containments (e.g., secondary containments for BWRs and shield buildings for PWRs that enclose the entire primary reactor containment or portions thereof) shall be subject to individual tests in accordance with the procedure established in the TS or associated bases.

The staff issued RAI 6.2-167 and RAI 15.4-26 to obtain information on leakage from the RB, test methods, and frequency of testing. In response, the applicant provided information on the test program and updated the DCD. The RB contaminated area, which serves as the safety envelope or, effectively, the secondary containment for release to the environment, is tested periodically under a positive pressure test as described in DCD Tier 2, Revision 9, Section 6.2.3, and ensures that the exfiltration will be less than the value assumed in the dose consequence analyses. The staff concludes that the test program meets the intent of 10 CFR Part 50, Appendix J, Option A.

RAI 6.2-167 and RAI 15.4-26 were being tracked as an open item. Based on the applicant's response, which included information on testing, RB leakage, and releases to the environment tied to DCD Revision 6 changes, the staff finds that these open items are resolved.

The staff issued RAI 6.2-168 to request clarification of issues concerning leakage from the RB. The RAI was based on DCD Revision 3. In response, the applicant provided information to address leakage rates from the RB. This information has been superseded by design changes and is no longer relevant. RAI 6.2-168 was being tracked as an open item, and it is now considered resolved.

6.2.3.4 *Conclusions*

The staff finds that the RB functional design, which provides for holdup in the contaminated portion (CONAVS) after an accident, and the subcompartment pressurization analysis are consistent with the guidance and criteria provided in SRP Sections 6.2.3 and 6.2.1.2 and other regulatory documents identified above. Thus, the design is acceptable.

6.2.4 Containment Isolation System

The containment isolation system (CIS) consists of isolation barriers, such as valves, blind flanges, and closed systems, and the associated instrumentation and controls required for the automatic or manual initiation of containment isolation. The purpose of the CIS is to permit the normal or postaccident passage of fluids through the containment boundary, while protecting against release to the environment of fission products that may be present in the containment atmosphere and fluids as a result of postulated accidents.

6.2.4.1 *Regulatory Criteria*

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations, in accordance with SRP Section 6.2.4, Rev. 3:

- GDC 1, as it relates to designing, fabricating, erecting, and testing safety-related SSCs to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, "Design bases for protection against natural phenomena," as it relates to designing safety-related SSCs to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform safety functions

- GDC 4, as it relates to designing safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and as it relates to the requirement that these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids

- GDC 16, as it relates to the requirement that reactor containment and associated systems establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment

- GDC 54, "Systems penetrating containment," as it relates to the requirement that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect their importance to safety and as it relates to designing such piping systems with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits

- GDC 55, "Reactor coolant pressure boundary penetrating containment," and GDC 56, "Primary containment isolation," as they relate to isolation valves for lines penetrating the primary containment boundary as parts of the RCPB (GDC 55) or as direct connections to the containment atmosphere (GDC 56) as follows:
  - One locked-closed isolation valve inside and one outside containment
  - One automatic isolation valve inside and one locked-closed isolation valve outside containment
  - One locked-closed isolation valve inside and one automatic isolation valve outside containment
  - One automatic isolation valve inside and one outside containment

- GDC 57, “Closed systems isolation valves,” as it relates to the requirement that lines that penetrate the primary containment boundary and are neither part of the RCPB nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment

- 10 CFR 52.47(b)(1), which requires that a design certification application 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 met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC’s regulations

- 10 CFR 52.47(a)(8) and 10 CFR 52.79(a)(17), as they relate to demonstrating compliance with any technically relevant portions of the requirements related to Three Mile Island (TMI) in 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv), for design certification and COL reviews, respectively

6.2.4.2 *Summary of Technical Information*

ESBWR DCD Tier 2, Revision 9, Section 6.2.4, describes the proposed CIS for the ESBWR. The CIS protects against releases of radioactive materials to the environment as a result of an accident.

The containment isolation function is accomplished by valves and control signals, required for the isolation of lines penetrating the containment. The CIS automatically closes fluid penetrations of fluid systems not required for emergency operation. Fluid penetrations supporting ESF systems have remote manual isolation valves that can be closed from the control room, if required.

DCD Tier 2, Revision 9, Table 6.2-13 identifies the RCPB influent lines, and DCD Tier 2, Revision 9, Table 6.2-14 identifies the RCPB effluent lines. DCD Tier 2, Revision 9, Tables 6.2-15 through 6.2-45 show the pertinent data for the containment isolation valves (CIVs). DCD Tier 2, Revision 9, Section 7.1.2 lists the criteria for the design of the leak detection and isolation system (LD&IS), which provides containment and reactor vessel isolation control. DCD Tier 2, Revision 9, Section 7.3.3 lists and explains the bases for assigning certain signals for containment isolation.

Power-operated CIVs have position-indicating switches in the control room to show whether the valve is open or closed. Power for valves used in series originates from physically independent sources without cross-ties to ensure that no single event can interrupt motive power to both closure devices.

CIV closure times are established by determining the isolation requirements necessary to keep radiological effects from exceeding the guidelines in 10 CFR 50.67. Chapter 15 discusses valve closure time bases for system lines, which can provide an open path from the containment to the environment. The design values of closure times for power-operated valves are more conservative than the above requirements.

Sensing instrument lines penetrating the containment follow all the recommendations of RG 1.11, “Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) issued March 1971 and Supplement to Safety Guide 11, Backfitting Considerations” issued February 1972. Each line has a 6-mm (1/4-in.) orifice inside the drywell, as close to the beginning of the instrument line as possible, and a manually operated isolation valve just outside the containment, followed by an excess flow check valve. The instrument line is designed such that the instrument response time is acceptable with the presence of the orifice and such that the flow restriction is not plugged.

The applicant stated that in general, the design of the CIS meets all requirements of GDC 54, 55, 56, and 57 and follows the guidance of RGs 1.11 and RG 1.141, "Containment Isolation Provisions for Fluid Systems (for Comment)," April 1978. DCD Tier 2, Revision 9, Section 6.2.4.3 gives a case-by-case analysis of all such penetrations. DCD Tier 2, Revision 9, Table 1.9-6 lists exemptions from the GDC.

The PCCS does not have isolation valves, as the heat exchanger modules and piping are designed as extensions of the safety-related containment. The design pressure of the PCCS is greater than twice the containment design pressure, and the design temperature is the same as the drywell design temperature.

Isolation valves, actuators, and controls are protected against damage from missiles. Tornado missile protection is afforded by the location of all CIVs inside the missile-proof RB. The arrangement of CIVs inside and outside the containment affords sufficient physical separation such that a high-energy pipe break would not preclude containment isolation. The CIS piping and valves are designed in accordance with seismic Category I standards.

CIVs and associated pipes are designed to withstand the peak calculated temperatures and pressures to which they would be exposed during postulated DBAs. They are designed in accordance with the requirements of ASME Code, Section III, and meet at least Group B quality standards, as defined in RG 1.26. The power-operated and automatic isolation valves will be cycled during normal operation to ensure their operability.

Redundancy is provided in all design aspects to satisfy the requirement that no single active failure of any kind should prevent containment isolation. Mechanical components are redundant, in that isolation valve arrangements provide backup in the event of accident conditions. Electrical redundancy is provided for each set of isolation valves to eliminate dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line are routed separately.

Plant operators will apply administrative controls by using established procedures and the checklist for all non-powered CIVs to ensure that their position is maintained and known. The position of all power-operated isolation valves is indicated in the control room. DCD Tier 2, Revision 9, Section 7.3.3 discusses instrumentation and controls for the isolation valves. DCD Tier 2, Revision 9, Section 6.2.6 discusses leak rate testing of isolation valves.

6.2.4.3 *Staff Evaluation*

The staff reviewed the description of the CIS using the review guidance and acceptance criteria of Section 6.2.4 of the SRP. SRP Section 6.2.4 identifies the staff’s review methodology and acceptance criteria for evaluating compliance with GDC related to those piping systems penetrating containment. During the review period, the applicant issued Revision 9 to DCD Tier 2. The staff finds that DCD Tier 2, Revision 9, Section 6.2.4, satisfies the guidance and acceptance criteria of Section 6.2.4 of the SRP.

The staff's review encompassed the following areas specified by Section 6.2.4 of the SRP and 10 CFR 50.34(f)(2)(xiv):

- CIS design, including the following:
  - The number and location of isolation valves (e.g., the isolation valve arrangements, location of isolation valves with respect to the containment wall, purge and vent valve conformance to SRP BTP 6-4, "Containment Purging During Normal Plant Operation" and instrument line conformance to RG 1.11)
  - The actuation and control features for isolation valves
  - The normal positions of valves and the positions valves take in the event of failures
  - The initiating variables for isolation signals and the diversity and redundancy of isolation signals
  - The basis for selecting closure time limits for isolation valves
  - The redundancy of isolation barriers
  - The use of closed systems as isolation barrier substitutes for valves
- The protection provided for CISs against loss of function caused by missiles, pipe whip, and natural phenomena
- Environmental conditions in the vicinity of CISs and equipment and their potential effect
- The mechanical engineering design criteria applied to isolation barriers and equipment
- The provisions for alerting operators of the need to isolate manually controlled isolation barriers
- Locating as close as practical
- Isolating at appropriate pressure
- Exceptions listed in DCD Tier 2, Revision 9, Table 1.9-6
- The provisions for, and TS pertaining to, operability and leak rate testing of isolation barriers
- The calculation of containment atmosphere released before isolation valve closure for lines that provide a direct path to the environs
- Containment purging and venting requirements of 10 CFR 50.34(f)(2)(xiv) and (xv)

Based on its review of the CIS as described in ESBWR DCD Tier 2, Section 6.2.4, the staff found that it needed additional information to resolve the open issues.

In RAI 6.2-102 and 6.2-102 S01, the staff requested additional information concerning the need for CIVs for the PCCS in accordance with the guidance in ANS-56.2/ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems." GEH responded that the design of the ESBWR containment cooling function does have precedent. In the Mark I style containment, the "light-bulb" shaped drywell is connected through a reinforced-concrete barrier by a series of metal ducts to the wetwell metal torus. This wetwell design is a contiguous part of the containment (not an extension or closed system outside of containment). This design contains features that are similar to those of the ESBWR, including the vent duct connections between the drywell and torus, which is a structural containment barrier that is not reinforced by concrete. The ESBWR containment is specifically designed to incorporate the safety-related function of containment cooling directly into the containment structure. Accordingly, GEH has pursued the development of a design that satisfies the applicable ASME Code, Section III, Div. 1, Subsection NE requirements for Class MC containment vessel design and construction.

According to DCD Tier 2, Revision 9, Section 6.2.2.4, the PCCS structural and leaktight integrity can be checked periodically by pressure testing. If additional ISI becomes necessary, ultrasonic testing (UT) could be performed during refueling outages. The scope and frequency of the inspections will be determined as part of the ISI program as stated in the ASME Code, Section XI.

GEH also considered the need for CIVs for the PCCS from a risk assessment perspective. GEH stated that the question of whether to install CIVs is a classic tradeoff between the following:

- The CIVs are automatically or manually closed before or during accidents involving fuel damage if one or more PCCS tubes and/or heat exchanger modules exhibit significant leakage.

- Inadvertent automatic (or manual) closure of multiple CIVs during any accident requiring successful operation of the PCCS condensers could result in inadequate containment heat removal and an increase in the core damage frequency and/or large release frequency (LRF).

For the first bullet above, it is not evident that instrumentation could be designed with sufficient reliability to correctly identify a significant radiological release from one or more tubes and to automatically close the associated CIVs to and from the PCCS heat exchanger module(s) without isolating intact modules. Depending on operators to manually close the CIVs would be an even less reliable approach.

For the second bullet above, the probabilistic risk assessment uses a containment heat removal success criterion of four of six PCCS loops. Thus, inadvertent isolation of three or more PCCS loops would defeat the function.

The staff performed a confirmatory calculation to assess the existing risk of the PCCS design without CIVs. GEH used a 72-hour mission time for calculating the probability of a heat exchanger leak, which is not conservative because it assumes that the only degradation mechanisms that could occur happen during the accident. While the staff acknowledges that the tubes are fabricated from corrosion-resistant material, they are not immune to all degradation mechanisms, and 2 years or more could elapse between test and inspection, depending on the final ISI program. The staff finds that the conservative 1×10^{-6}/h heat exchanger leakage rate (i.e., probability per unit time of a leak) compensates for this nonconservative assumption. Finally, the staff used six PCCS heat exchanger modules in its analysis.

The staff used the following inputs when it repeated the risk assessment:

- A total core damage frequency (internal and external events at power) of 2.3×10-8/year (yr) rather than the GEH value of 5.81×10-9/yr

- A standby failure rate of 3×10-8/h for large heat exchanger leaks (where the leak is greater than 0.19 m3 per minute [50 gpm]) from NUREG/CR–6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” issued February 2007, which is the most recent and generally accepted operating experience data source

- A fault exposure time of T/2, where T = 8,760 hours (i.e., assuming 2 years between testing/inspection).

- A total of six PCCS heat exchange modules

The existing LRF from the proposed design (without CIVs) was recalculated as

$$(2.3\times10^{-8}/\text{yr}) * (3\times10^{-8}/\text{h}) * (8{,}760\ \text{h}) * 6 = 3.6\times10^{-11}/\text{yr for LRF.}$$

This value is 2 orders of magnitude greater than the GEH estimate of 4×10^{-13}/yr for the existing level of risk from large release due to PCCS leakage during severe accidents. However, the value of 3.6×10^{-11}/yr remains very low compared to the existing LRF from all other at-power severe accidents of about 1.7×10^{-9}/yr. More importantly, it remains lower by 4 or more orders of magnitude than the potential LRF increase in the alternate design due to inadvertent isolation of three or more PCCS heat exchanger modules during accidents requiring containment heat removal.

The staff’s evaluation confirms the applicant’s risk assessment conclusions and provides reasonable assurance that the proposed PCCS design without isolation valves represents lower risk than the alternative design with isolation valves.

The staff finds that the PCCS provides a functional feature of the ESBWR primary containment that ensures cooling in the event of a DBA. In addition, the PCCS provides an inherent capability designed into the containment structure, and is not a separate fluid process system. This is a specific departure from past BWR plant designs. All previous BWR containment designs have relied on an external, pressurized, active fluid heat exchange system to provide containment cooling in response to a DBA. The PCCS negates the need for a separate, active safety-related cooling system and thus eliminates the need for fluid piping penetrations.

RAI 6.2-102 was being tracked as an open item in the SER with open items. Based on the above review and the precedent of the Mark I containment example, the staff finds the proposed design of the PCCS without isolation valves acceptable. RAI 6.2-102 is resolved.

In RAI 6.2-103, the staff asked that DCD Tier 2, Table 1.9-6, be revised to state that the PCCS differs from SRP Section 6.2.4 acceptance criteria, in that it has no CIVs. RAI 6.2-103 was being tracked as an open item in the SER with open items. The applicant indicated that it described its position on PCCS isolation in response to RAI 6.2-102 S01 and the issue was resolved under that RAI. This staff concern in RAI 6.2-103 is resolved by the response to RAI 6.2-102, which concluded that the proposed design of the PCCS does not require CIVs and does not deviate from SRP Section 6.2.4 acceptance criteria.

The staff also asked that the process radiation monitoring system be added to DCD Tier 2, Table 1.9-6, because it has both CIVs outside containment. The applicant responded that these lines conform to the provisions of RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment," Revision 1, March 2010. (as described in its response to RAI 6.2-127), which would mean that the lines do conform to SRP Section 6.2.4 acceptance criteria.

However, the applicant had not demonstrated that the system does conform to RG 1.11 (See RAI 6.2-127 S01), and so the staff requested that the applicant add the process radiation monitoring system to Table 1.9-6 or change its design to bring it into conformance with SRP Section 6.2.4. The applicant responded that it would address its position on containment isolation provisions of the process radiation monitoring system as part of its response to RAI 6.2-127 S01. This staff concern is resolved with the closure of RAI 6.2-127 S01, because the applicant revised the DCD to include both inboard and outboard CIVs.

Based on the applicant’s response, RAI 6.2-103 is resolved.

In RAI 6.2-104, the staff pointed out that four systems did not meet the specific requirements of GDC 55 and 56. DCD Tier 2, Revision 3, Table 1.9-6, listed three of the systems, and the fourth was the PCCS. The staff asked the applicant to clarify or correct this apparent discrepancy. RAI 6.2-104 was being tracked as an open item in the SER with open items. To correct the inconsistency, the applicant responded that in DCD Tier 2, Section 6.2.4, Revision 5, it had added a statement that there are exceptions to the explicit requirements of GDC 55 and 56 and that these exceptions are listed in Table 1.9-6 and are qualified on a case-by-case basis. Based on the applicant’s response, RAI 6.2-104 is resolved.

In RAI 6.2-106, the staff requested that the third bullet in DCD Tier 2, Section 6.2.4.1 be revised to remove the statement “to the greatest extent practicable consistent with safety and reliability.” As applicable, the applicant should request an exemption, or revise the statement to include “except as noted below” and then provide the specific exceptions. RAI 6.2-106 was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Revision 5, Section 6.2.4.1, third bullet, to remove the statement identified above, added a reference to identify the exemptions to the explicit requirements of GDC 55 through 57, and identified these exemptions in DCD Tier 2, Table 1.9-6. Based on the applicant’s response, RAI 6.2-106 is resolved.

RAI 6.2-107 requested that the applicant clarify the following statement in DCD Tier 2, Section 6.2.4.1, seventh bullet: “Containment isolation valves and associated piping and penetrations meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, or MC, in accordance with their quality group classification.” Class MC does not appear to meet the guidelines for a CIS. RAI 6.2-107 was being tracked as a confirmatory item in the SER with open items. In response, the applicant stated that the seventh bullet refers to the code for the piping (ASME Section III, Class 1 or 2), as well as the steel components (ASME Section III, Class MC) of other than piping penetrations. In response to a supplement request, GEH revised DCD Tier 2, Revision 5, Section 6.2.4.1, seventh bullet, to clarify that CIVs and associated piping meet the requirements of ASME Code Section III, Class 1 or 2, in accordance with their quality group classifications and added another bullet stating that piping penetrations (that is, penetrations themselves and not the pipes) are designed to the requirements of Subsection NE (MC components) of Section III of the ASME Code.

The staff confirmed that this change was included in DCD Tier 2, Revision 5. Based on the applicant’s response, RAI 6.2-107 is resolved.

In RAI 6.2-109, the staff requested information about CIV closure times. In DCD Revision 3, the applicant made appropriate revisions and included acceptable CIV closure times in DCD Tier 2, Tables 6.2-16 through 6.2-42, except as follows:

- Isolation Condenser System—In DCD Tier 2, Revision 3, Tables 6.2-24, 6.2-26, 6.2-28, and 6.2-30, 20-mm (0.8–in.) CIVs have closure times of 30 seconds or less.

- High-Pressure Nitrogen Gas Supply System—In Table 6.2-40, 50-mm (2–in.) CIVs F0009 and F0026 have closure times of 30 seconds or less.

Because DCD Tier 2, Revision 3, Section 6.2.4.2.1, states that CIVs that are 80 mm (3 in.) or less in diameter "generally close within 15 seconds," consistent with national standard ANS-56.2/ANSI N271-1976, Section 4.4.4, the staff was unsure if the quoted closure times of "30 seconds or less" for the above two systems are correct. RAI 6.2-109 was being tracked as an open item in the SER with open items.

The applicant responded that it changed the closure times for the CIVs for the isolation condenser and high-pressure gas supply systems as listed in DCD Tier 2, Revision 5, Tables 6.2-24, 6.2-26, 6.2-28, 6.2-30, and 6.2-40, to indicate that the valves close within 15 seconds. Based on the applicant's response, RAI 6.2-109 is resolved.

In RAI 6.2-110, the staff questioned whether the instrument lines in the ESBWR design conform to the provisions of RG 1.11. RAI 6.2-110 was being tracked as an open item in the SER with open items. GEH stated that it had revised the first paragraph of DCD Tier 2, Revision 5, Section 6.2.4.2.2, to include sufficient information demonstrating conformance to each of the specific regulatory positions of RG 1.11, for every instrument line. Based on the applicant's response, RAI 6.2-110 is resolved.

In RAI 6.2-115(B), the staff asked for a more complete discussion of the single-failure evaluations performed for the CIS. In response the applicant stated that it would revise DCD Tier 1, Revision 5, Section 2.15.1 and Table 2.15.1-2, and DCD Tier 2, Revision 5, Section 6.2.4.3.3, as shown in attached markups. GEH stated that the single-failure evaluation method for containment penetration isolation designs is based on the commitment to standards ANSI/ANS 58.9, "Single Failure Criteria for LWR Safety-Related Fluid Systems," and Institute of Electrical and Electronic Engineers (IEEE) 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems—Description" (see DCD Tier 2, Table 1.9-22), and RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems," Revision 2, November 2003 (see DCD Tier 2, Tables 1.9-21 and 7.1-1, and Sections 7.13.3 and 7.5.2). DCD Tier 2, Section 6.2.4.3.3, clarifies the method by which single failure is evaluated for containment isolation. Those commitments will be demonstrated under DCD Tier 1, ITAAC Table 2.15.1-2.

RAI 6.2-115 was being tracked as an open item in the SER with open items. The staff has reviewed the applicant's response regarding the single-failure evaluations for the CIS and concluded it meets the requirements of RG 1.53 and national standard ANSI/ANS 58.9 and is therefore acceptable. Based on the applicant's response, RAI 6.2-115 is resolved.

In RAI 6.2-117, the staff requested that more detailed information be added to DCD Tier 2, Section 6.2.4.2.5, to describe the administrative controls to the extent that they are required by the regulations. RAI 6.2-117 was being tracked as a confirmatory item in the SER with open items. In response, the applicant revised DCD Tier 2, Revision 5, Section 6.2.4.2.5, to describe the manual valves that can be configured only to permit administrative control. Compliance with GDC 55 through 57 requires that the manual CIVs be locked closed. The staff has reviewed the applicant's response and finds it acceptable as these administrative controls meet the requirements of RG 1.141 and satisfy the national standards of ANS-56.2/ANSI N271-1976.

The staff confirmed that this change was included in DCD Tier 2, Revision 5. Based on the applicant's response, RAI 6.2-117 is resolved.

The containment isolation provisions of the IC condensate, venting, and purge lines consist of one barrier (a closed system) outside containment and two CIVs inside containment. In RAI 6.2-119 S01, the staff stated that this design does not comply with the explicit requirements of GDC 55 or GDC 56 and is inconsistent with the appropriate guidance documents (i.e., SRP Section 6.2.4, Revision 2; RG 1.141; and national standard ANS-56.2/ANSI N271-1976) concerning alternate means for complying with GDC 55 or GDC 56. These GDC allow alternate isolation provisions, other than their explicit requirements, if "it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis." RAI 6.2-119 S01 was being tracked as open items in the SER with open items.

The applicant stated in response, that because of the physical arrangement of the ICS condensate, venting, and purge line piping, it is impractical to locate an isolation valve outside the containment boundary. Such a valve would be under water and therefore inaccessible and less reliable than a valve located inside the containment boundary. As an alternative, two CIVs in series are located inside containment as close as possible to the containment boundary. The piping between the valves and containment boundary is designed to meet conservative requirements, precluding the occurrence of breaks in these areas. The ICS piping and components outside containment form a closed system designed to withstand the full reactor pressure.

The staff finds that in addition to the explicit GDC 55 and 56 configuration of one CIV inside and one outside containment, the guidance documents allow two other configurations: (1) one CIV and a closed system, both outside containment, or (2) two CIVs outside containment. The ICS design does not conform to either of these. The NRC has the authority to approve additional isolation configurations under the "other defined basis" provision of the GDC, but the applicant must adequately justify its proposed alternative to ensure sufficient safety, consistent with the overall containment isolation design philosophy expressed in the GDC and guidance documents. For example, SRP Section 6.2.4 states, "If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment." In the ICS case, locating a CIV outside containment would place it under water all of the time. This is sufficient justification for moving it inside containment.

Based on the above evaluation, the staff finds that the containment isolation design for the ICS is considered an adequate alternative to the requirements of GDC 55 because a single failure would not disable the containment isolation function. Therefore, RAI 6.2-119 is considered resolved. Based on the applicant's response, RAI 6.2-119 S01 is resolved.

RAI 6.2-121 is subsidiary to RAI 6.2-119. In RAI 6.2-119 and RAI 6.2-121, the staff made similar requests regarding the containment isolation design for the ICS. The containment isolation provisions of the isolation condenser condensate, venting, and purge lines consist of one barrier (a closed system) outside containment and two CIVs inside containment. The first

RAI concerned the influent lines and the second RAI concerned the effluent lines. RAI 6.2-119 S01 addressed both the influent and effluent lines of the system. Based on the applicant's acceptable response to RAI 6.2-119 S01, RAI 6.2-121 is resolved.

In RAI 6.2-120, the staff noted that DCD Tier 2, Revision 1, Section 6.2.4.3.1.2, under the heading describes the power-operated main steam isolation valves (MSIVs) as closing under either spring force or gas pressure. The staff questioned this statement, considering that virtually every BWR main steam isolation valve (MSIV) in the United States needs both gas pressure and spring force to close under accident conditions.

The applicant's response to RAI 6.2-120 explained the operation of the valves, which is similar to the operation of the MSIVs in other BWRs. RAI 6.2-120 was being tracked as an open item in the SER with open items. The response included a proposed DCD Revision 3, Section 6.2.4.3.1.2. However, the applicant did not incorporate the proposed revision in DCD Revision 3, Section 6.2.4.3.1.2. On another note, the RAI response and DCD version refer to DCD Section 5.4.5 for further information, but that section does not address this particular issue. In RAI 6.2-120 S01 the staff requested the revision of the DCD to include the appropriate information as presented in the proposed DCD Revision 3 and to revisit the reference to Section 5.4.5.

The applicant's response to RAI 6.2-120 S01 stated that DCD Tier 2, Section 5.4.5, is the correct location for information regarding the design requirements and functional evaluation of the MSIVs, including the description of all relevant forces to which the actuation mechanism must respond during normal or abnormal operating conditions. The applicant provided the revised markup of DCD Tier 2, Section 5.4.5, instead of revising Section 6.2.4.3.1.2. Based on the above review, the staff finds this acceptable. Based on the applicant's response, RAI 6.2-120 is resolved.

In RAI 6.2-122, the staff requested that information about the containment isolation design for the FAPCS be provided in Section 6.2.4.3.2 to support the deviation from GDC 56. The staff also indicated that DCD Tier 2, Table 6.2-33b should be corrected to be consistent with Table 6.2-33a for the CIV position on loss of electric or air supply. In response, GEH corrected DCD Tier 2, Table 6.2-33b, to be consistent with DCD Tier 2, Table 6.2-33a, for CIV position in case of power failure.

GEH also revised DCD Tier 2, Section 6.2.4.3.2 for the FAPCS to provide the following information.

> The lines from the FAPCS penetrate the containment separately and are connected to the drywell spray, the suppression pool, the GDCS pools, and the reactor well drain.
>
> The reactor well drain line contains two manual valves inside the containment that are locked closed during normal operation. This arrangement is an exception to GDC 56, which requires that such lines contain one isolation valve outside and one isolation valve inside the containment. The alternative arrangement with both valves inside containment is necessary because a valve outside containment would be submerged in the reactor well, making it inaccessible and less reliable. The isolation valves are located as close as possible to the containment, and the piping between the outermost valve and the

containment boundary is designed to conservative requirements to preclude breaks in this area.

In each of the remaining influent lines, there is one pneumatic-operated or equivalent-shutoff valve outside and one check valve inside the containment. Only the GDCS pool return line pneumatic-operated or equivalent-shutoff valve is automatically closed on a containment isolation signal.

Before it exits containment, the FAPCS suction line from the suppression pool branches into two parallel lines, each of which penetrates the containment boundary. Once outside, each parallel flow path contains two pneumatic isolation valves in series, after which the lines converge into a single flow path. The CIVs are normally closed and fail as-is for improved reliability. "Fail-as-is" valves are acceptable because the valves are normally closed, will only be open when it is necessary to provide cooling to the suppression pool, and do not communicate with the drywell atmosphere. This arrangement is an exception to GDC 56, which requires that such lines contain one isolation valve outside and one isolation valve inside the containment. Such an alternative arrangement is necessary because the inboard valve could potentially be under water under certain accident conditions. Leak detection is provided for CIVs on the suppression pool suction line, and valves are located as close as possible to the containment.

The CIVs on the FAPCS suppression pool suction and return lines are considered to fail in the position of greatest safety. The CIVs in the suppression pool supply and return lines are closed for all normal operating conditions, except for temporary usage when suppression pool cooling or cleaning is needed. However, if the suppression pool cooling mode has been initiated before an accident, then it is more desirable to continue removing decay heat than to terminate the mode and isolate the system. This is clarified in DCD Tier 2, Revision 5, Section 6.2.4.3.2. Therefore, the fail-as-is feature allows these valves to remain in an open position, which provides additional reliability for the RTNSS functions of suppression pool cooling and LPCI. Furthermore, the CIVs are designed to accommodate a single failure such that the line can still be isolated with the loss of a single division of power.

While the functions of suppression pool cooling and LPCI are not considered ESFs, they are considered RTNSS backups to ESFs, including the PCCS and GDCS. Therefore, the regulatory treatment that has been assigned to these functions, which utilize the FAPCS suppression pool flow path, is justification for using the provisions of SRP Section 6.2.4, Revision 2, Section II.6.d.

The staff reviewed the information provided by GEH in response to RAI 6.2-122 as indicated above. The staff found that GEH provided the required information about the containment isolation design for the FAPCS in DCD Tier 2, Section 6.2.4.3.2 to support the deviation from GDC 56 as per guidelines of SRP Section 6.2.4. RAI 6.2-122 was being tracked as an open item in the SER with open items. Based on the applicant's response, RAI 6.2-122 is resolved.

In RAI 6.2-123, the staff noted that, for the influent and effluent lines of the containment inerting system, described in DCD Tier 2, Revision 1, Sections 6.2.4.3.2.1 and 6.2.4.3.2.2, all of the CIVs were outside of containment, but without adequate justification as described in the guidelines of SRP Section 6.2.4, Revision 2 (Section II.d), RG 1.141, and national standard ANS-56.2/ANSI N271-1976 (Sections 3.6.5 and 3.7). RAI 6.2-123 was being tracked as an open item in the SER with open items. The applicant's response provided changes to the DCD that address the guidelines.

The DCD states that the penetration of the containment inerting system consists of two tandem quarter-turn or equivalent shutoff valves (normally closed), in parallel with two tandem stop or shutoff valves. All isolation valves on these lines are outside of the containment so that they are not exposed to the harsh environment of the wetwell and drywell and are accessible for maintenance, inspection, and testing during reactor operation. Both CIVs are located as close as practical to the containment. The valve nearest to the containment has the capability to detect and terminate a leak. The piping between the containment and the first isolation valve and the piping between the two isolation valves are designed to meet the requirements of SRP Section 3.6.2. The piping is designed to meet Safety Class 2 and seismic Category I design requirements and to withstand the containment design temperature, design pressure, and LOCA transient environment and is protected against an HELB outside containment when needed for containment isolation.

The staff has reviewed the applicant's response and redundant CIV arrangement. Because (1) the containment inerting isolation valves are normally closed during reactor operation, (2) piping between the containment and the CIVs is conservatively designed to preclude a breach of piping integrity, and the design of the valve and/or piping compartment provides the capability to detect leakage from the valve shaft and or bonnet seals and terminate the leakage according to the requirements of SRP Sections 3.6.2 and 6.2.4, and (3) locating both CIVs outside containment protects the valves from the harsh environment of the wetwell and drywell and allows accessibility for inspection and testing, the staff finds acceptable the proposed location of both inerting system CIVs outside the containment. Based on the applicant's response, RAI 6.2-123 is resolved.

RAI 6.2-125 is subsidiary to RAI 6.2-122. In RAI 6.2-122 and RAI 6.2-125, the staff made similar requests regarding the containment isolation design for the FAPCS. The first RAI concerned the influent lines and the second RAI concerned the effluent lines. RAI 6.2-122 S01 addressed both the influent and effluent lines of the system. Based on the applicant's acceptable response to RAI 6.2-122 S01, RAI 6.2-125 is resolved.

In RAI 6.2-127, the staff questioned the design of the process radiation monitoring system, particularly the placement of all CIVs outside of containment. RAI 6.2-127 was being tracked as an open item in the SER with open items. The applicant responded that the lines 1 in. (25 mm) in diameter should be treated as instrument lines and that the design is acceptable because it follows the guidance in RG 1.11, Revision 1. The staff asked the applicant to provide a discussion showing that these lines conform to RG 1.11, or, if not, to identify the requirements for non-instrument lines.

In response to RAI 6.2-127 S01, the applicant stated that the design has been changed to include an inboard and outboard CIV on penetrations for the fission products monitor sampling line and return line. These two isolation valves are designed to a fail-as-is condition. In DCD Tier 2, Revision 5, the applicant added a new Figure 6.2-30 to show these isolation valves and revised DCD Tier 2, Tables 3.9-8 and 6.2-42 to include both inboard and outboard CIVs. Based on the acceptable applicant's response, RAI 6.2-127 and the supplement S01 are resolved.

In RAI 6.2-128, the staff noted that DCD Tier 2, Revision 1, Tables 6.2-39 through 6.2-42, does not include information covering the chilled water, high-pressure nitrogen gas supply, and process radiation monitoring systems. RAI 6.2-128 was being tracked as an open item in the

SER with open items. In DCD Tier 2, Revision 3, the applicant filled in the tables for the above systems. Based on its review, the new information was generally acceptable, but the staff had the following questions:

A. For the Chilled Water and High Pressure Nitrogen Gas Supply Systems, the stated applicable basis is GDC 57. The applicant's revised response to RAI 6.2-129 recognizes that no ESBWR system credits a closed system inside containment (per GDC 57) as a containment isolation barrier. Please correct the tables in the DCD.

B. For the High Pressure Nitrogen Gas Supply and Process Radiation Monitoring Systems, the tables indicate that DCD Tier 2 figures for the systems are "N/A." Why are system figures not applicable? When will figures be provided?

C. Closure times for CIVs in the High Pressure Nitrogen Gas Supply System are unacceptable. See RAI 6.2-109 S01 for details.

In response, the applicant stated the following:

> (A) These tables for the Chilled Water System (CWS) and High Pressure Nitrogen Gas Supply System (HPNSS) were corrected in DCD Tier 2, Revision 4, to indicate GDC 56 as the applicable basis; (B) For the HPNSS, Table 6.2-40 will be revised to reference the appropriate DCD Tier 2 figures. For the Process Radiation Monitoring System, the response to RAI 6.2-127 S01 provides the appropriate DCD Tier 2 changes in Revision 5. For CIVs in the High Pressure Nitrogen Gas Supply System, response to supplement RAI 6.2-109 provides acceptable closure times.

The staff finds the applicant has provided the required information for the CWS, HPNSS and process radiation monitoring system CIVs in the DCD as per GDC 56. Based on the applicant's response, RAI 6.2-128 is resolved.

In RAI 6.2-131, the staff requested that the applicant discuss the following in the DCD:

A. The automatic isolation signals for CIVs and their diversity of parameters sensed, per item II.I of SRP Section 6.2.4, Revision 2.

B. Classification of systems as essential or non-essential and automatic isolation of non-essential systems during an accident per item II.h of SRP Section 6.2.4, Revision 2, and item II.E.4.2 of NUREG–0737.

C. Reducing the containment setpoint pressure that initiates containment isolation for non-essential penetrations to the minimum compatible with normal operating conditions, per item II.k of SRP Section 6.2.4, Revision 2, and item II.E.4.2 of NUREG–0737.

The GEH responses to parts A and B of RAI 6.2-131 are acceptable. In response to part A, GEH stated that DCD Tier 2, Subsections 5.2.5 and 7.3.3.2 provide a discussion of the automatic isolation signals for CIVs and their diversity of parameters sensed as per item II.I of SRP Section 6.2.4, Revision 2. DCD Tier 2, Subsection 6.2.4 was revised to include a reference to the discussions in Subsection 5.2.5 and 7.3.3.2. The staff evaluation finds the response to part A acceptable.

In response to part B, GEH stated that instead of terms ‘essential’ or ‘nonessential’ for the classification of systems, GEH used the terms ‘safety-related’ and ‘nonsafety-related’ for clarity when describing the importance of the functions of a system with regard to safety, similar to the terminology in NUREG–0737, “Clarification of TMI Action Plan Requirements,” issued November 1980, item II.E.4.2, Table 1A-1. DCD Tier 2, Subsection 6.2.4.1, provides the criteria for categorizing the fluid penetrations that require automatic isolation verses remote manual containment isolation based on the same basic criteria further described in SRP Section 6.2.4 Revision 3, Item II.8. DCD Section 6.2.4.1 states, “The containment isolation function automatically closes fluid penetrations of fluid systems not required for emergency operation. Fluid penetrations supporting ESF systems have remote manual isolation valves that can be closed from the control room, if required.” DCD Section 6.2.4.2 describes the systems containing penetrations that support or provide a flow path for emergency operation of ESF systems not automatically isolated. The staff evaluation finds the GEH response to part B acceptable.

However, the staff had a further request for part C. In RAI 6.2-131 S01, part C, GEH proposed a change to DCD Tier 2, Appendix IA, to include the following:

> The alarm and initiation setpoints of the LD&IS are set to the minimum compatible with normal operating conditions to initiate containment isolation for containment penetrations containing process lines that are not required for emergency operation. The values for these setpoints are determined analytically or are based on actual measurements made during startup and preoperational. In a supplement [to] this RAI, the staff requested that if setpoints are to be determined analytically, provide the actual numerical value and justify that it is minimum compatible with normal operating conditions. If the setpoints are to be based on actual measurements during startup and preoperational tests then revise the DCD to provide more details regarding how and when this setpoint will be determined.

GEH also stated that the ESBWR is in compliance with NUREG–0737. As currently stated in DCD Tier 2, Appendix 1A, Table 1A-1, Item II.E.4.2, the alarm and initiation setpoints for a high-drywell-pressure condition are reduced to the minimum values compatible with normal operating conditions for containment penetrations containing process lines that are not required for emergency operation. However, the primary concern is to ensure that the high-drywell-pressure setpoint is set conservatively to the analytical limit used in the safety analyses. To clarify the basis of the high-drywell-pressure initiation signal, DCD Tier 2, Appendix 1A, Table 1A-1, Item II.E.4.2, will be revised to state that the high-drywell-pressure setpoint is based on the analytical limit used in the safety analyses, and the reference to startup and preoperational test measurements will be deleted. The staff reviewed the proposed changes in DCD Tier 2, Revision 6, and finds them acceptable.

The value for the high–drywell-pressure setpoint is the same for both the reactor protection system (RPS) scram signal and the containment isolation signal. DCD Tier 2, Revision 9, Table 6.2-2, shows the analytical limit for the high-drywell-pressure signal as 13.8 kPaG (2 psig). This value is an upper analytical limit and is the basis for a setpoint calculation that will be performed to determine the actual instrument setting. This setpoint calculation will be based on the GEH setpoint methodology (see NEDE-33304P, "GEH ESBWR Setpoint Methodology"

Revision 4, dated May, 2010). A setpoint based on this analytical limit is compatible with the maximum normal operating drywell pressure of 8.96 kPaG (1.3 psig) identified in DCD Tier 2, Chapter 16. The analytical limit is sufficiently low to ensure the performance of the necessary safety actions and, at the same time, high enough not to cause spurious reactor trips. The alarm and initiation setpoints of the LD&IS are set as low as compatible with normal operation.

The actual setpoint will be based on instrument sensitivity and tolerance relating to actual installed instrument type, instrument range, setpoint drift, post-event function time, and environmental and process conditions and will ensure that the analytical limit is met. DCD Tier 2, Revision 9, Sections 5.2.5 and 7.3.3, discuss the LD&IS parameters used to initiate these signals.

Based on the above evaluation, the staff finds the GEH response to RAI 6.2-131 S01 acceptable. Based on the applicant's response, RAI 6.2-131 is resolved.

DCD Tier 2, Revision 3, contained a new table, Table 6.2-47. The staff compared this table with Tables 6.2-15 through 6.2-42, which were to provide "pertinent data for the containment isolation valves" (See DCD Tier 2, Revision 3, Section 6.2.4.2), presumably in a comprehensive way. However, Table 6.2-47 included many containment piping penetrations (i.e., approximately 122) that were not covered in Tables 6.2-15 through 6.2-42 or elsewhere in DCD Tier 2, Revision 3, Section 6.2.4. Further, Table 6.2-47 contained virtually no information on the containment isolation provisions for these lines, other than incomplete information on leakage rate testing. Some systems were not covered in Tables 6.2-15 through 6.2-42.

In RAI 6.2-157 the staff requested that GEH address this issue. RAI 6.2-157 was being tracked as an open item in the SER with open items.

In response, GEH revised DCD Tier 2, Table 6.2-47 to contain the required information for containment penetrations subject to Type A, B, and C testing and satisfies SRP Section 6.2.4 criteria. The CIV information in DCD Tier 2, Tables 6.2-15 through 6.2-45 was also revised and information was added on the isolation valves in the makeup water system, service air system, containment monitoring system, and equipment and floor drain system.

In RAI 6.2-157 S01 the staff stated that COL Information Item 6.2-1-H in DCD Tier 2, Section 6.2.8 requires the Licensee to provide the missing information in Tables 6.2-16 through 6.2-45. This is the length of pipe between the containment and the isolation valve(s). Although it is understood that this information is not available until detailed design, GEH should provide acceptance criteria such that this information can be validated in ITAAC.

In response, GEH committed to the following design requirements:

> The containment isolation valves shall be located as close to the containment as practical. Sufficient space shall be provided between the valves and containment boundary to permit the following:
>
> - In-service inspection of non-isolable welds
> - Appendix J of 10 CFR Part 50 leak testing
> - Cutout and replacement of isolation valves using standard pipe fitting tools and equipment

- Local control

- Valve seat resurfacing in place

In RAI 6.2-157 S02 the staff stated that the proposed design criteria for locating the pipes is reasonable. However, the GEH response did not allow a safety conclusion that the ESBWR complies with GDC 55, 56, and 57. Therefore, GEH must include the appropriate design in the DCD to demonstrate compliance with GDC 55, 56, and 57, and an ITAAC item must also be added to ensure that the detailed design complies with the guidance in the DCD.

In response, GEH stated that the design considerations for locating CIVs as close to the containment as practical, which were provided in the response to RAI 6.2-157 S01, would be added to DCD Tier 2, Section 6.2.4.2. An ITAAC item would be added to DCD Tier 1, Table 2.15.1-2, to document the location of CIVs relative to containment and to review these locations relative to the design considerations. COL Information Item 6.2-1-H, which was to provide the pipe lengths between the CIVs and containment, would be deleted from DCD Tier 2, Section 6.2.8. The piping lengths in DCD Tier 2, Tables 6.2-16 through 6.2-45, would also be deleted.

DCD Tier 1, Section 2.15.1 and Table 2.15.1-2, and DCD Tier 2, Sections 6.2.4.2 and 6.2.8 and Tables 6.2-16 through 6.2-45, were to be revised accordingly. The staff confirmed that these changes were incorporated in DCD Tier 1 and 2, Revision 6.

Based on the applicant’s response, RAI 6.2-157 and RAIs 6.2-157 S01-S02 and the associated open items are resolved.

In DCD Tier 2, Revision 5, Tables 6.2-36, 6.2-37, and 6.2-38 refer to Figure 9.4-14 for valve location. However, in Revision 5, Figure 9.4-14 was moved to Chapter 6. In RAI 6.2-199, the staff requested that the applicant update the above tables to reflect the proper reference and update Figure 6.2-29 to include the containment inerting system. In addition, Figure 6.2-29 should include isolation valve F023 and penetration numbers.

GEH agreed to make the necessary changes in DCD Tier 2, Tables 6.2-36, 6.2-37, and 6.2-38 and Figure 6.2-29. The staff confirmed that these changes were incorporated in DCD Tier 2, Revision 6. Based on the applicant’s response, RAI 6.2-199 is resolved.

In DCD Tier 2, Revision 5, Tables 6.2-16 to 6.2-40 present CIV design information. These tables typically refer to other Tier 2 figures for information such as isolation valve(s) and containment penetration. However, many of the referenced figures do not show such information. In tables that refer to other figures for design details, the referenced figures should be updated to show the isolation valve(s) and penetration numbers.

In Tables 6.2-41, 6.2-43, 6.2-44, and 6.2-45, the entries that typically give design information show “N/A” for Tier 2 figures. Thus, there is no design figure (e.g., piping and instrumentation diagram, process diagram). In RAI 6.2-200, the staff requested that these tables be revised to include figure(s) showing the isolation valve(s) and penetration numbers.

In response, GEH stated that it would revise the DCD to ensure that there are figures showing all CIVs, and that all CIVs and penetrations are labeled with their component numbers on the figures. GEH also agreed to make additional changes to the DCD to correct information associated with CIVs.

GEH provided a markup of the revised tables and figures. The staff confirmed that these changes were incorporated in DCD Tier 2, Revision 6. Based on the applicant's response, RAI 6.2-200 is resolved.

Generic Issues

The two generic issues included in the staff's review of the CIS are TMI Action Plan Items II.E.4.2, "Containment Isolation Dependability," and II.E.4.4, "Containment Purging During Reactor Operation" of NUREG–0737.

II.E.4.2, "Containment Isolation Dependability" (10 CFR 50.34(f)(2)(xiv))

The governing regulation, 10 CFR 50.34(f)(2)(xiv), states the following:

> Provide containment isolation systems that: (II.E.4.2)
>
> A. Ensure all non-essential systems are isolated automatically by the containment isolation system,
>
> B. For each non-essential penetration (except instrument lines) have two isolation barriers in series,
>
> C. Do not result in reopening of the CIVs on resetting of the isolation signal,
>
> D. Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,
>
> E. Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

DCD Tier 2, Revision 9, Table 1A-1, states that the ESBWR CIS meets the NRC requirements, including the post-TMI requirements. In general, this means that two barriers are provided, as discussed in DCD Tier 2, Revision 9, Section 6.2.4.3.

Redundancy and physical separation are required in the electrical and mechanical design of the CIS to ensure that no single failure in the system prevents it from performing its intended functions. Electrical redundancy is provided for each set of isolation valves, such that the unavailability of any two safety-related electrical divisions will not prevent isolation from occurring. Electrical cables for isolation valves in the same line are routed separately. Cables are selected and based on the specific environment to which they may be subjected (e.g., magnetic fields, high radiation, high temperature, and high humidity).

Safety-related or nonsafety-related (essential or nonessential) classification of SSCs for the ESBWR design is addressed in DCD Tier 2, Revision 9, Section 3.2 and identified in DCD Tier 2, Revision 9, Table 3.2-1. Section 3.2 also presents the basis for classification.

The CIS, in general, closes fluid penetrations for support systems that are not safety-related. The design of the control systems for automatic CIVs ensures that resetting the isolation signal does not result in the automatic reopening of CIVs.

Actuation of the CIS is automatically initiated by the LD&IS, at specific limits (described in DCD Tier 2, Revision 9, Sections 5.2.5 and 7.3.3) defined for reactor plant operation. The LD&IS is designed to detect, monitor, and alarm leakage inside and outside the containment and automatically initiates the appropriate protective action to isolate the source of the leak. Various plant variables are monitored, including pressure, and these are used in the logic to isolate the containment. The drywell pressure is monitored by four divisional channels, using pressure transmitters to sense the drywell atmospheric pressure from four separate locations. A pressure rise above the nominal level indicates a possible leak or loss of reactor coolant within the drywell. A high-pressure indication is alarmed in the main control room (MCR) and initiates reactor scram and, with the exception of the MSIVs, closure of the CIVs in certain designated process lines.

All ESBWR containment purge valves meet the criteria provided in SRP BTP 6-4, "Containment Purging During Normal Plant Operation." The main purge valves are fail-closed and are verified to be closed at a frequency interval of 31 days as defined in the plant TS (SR 3.6.13.1). All purge and vent valves are pneumatically operated, fail closed, and receive containment isolation signals. Bleed valves and makeup valves can be manually opened remotely in the presence of an isolation signal, by utilizing override control if continued inerting is necessary.

In the ESBWR design, redundant primary CIVs (purge and vent) close automatically upon receipt of an isolation signal from the LD&IS. The LD&IS is a four-division system designed to detect and monitor leakage from the RCPB and, in certain cases, isolates the source of the leak by initiating closure of the appropriate CIVs. Various plant variables are monitored, including radiation level, and these are used in the logic to initiate alarms and the required control signals for containment isolation. High-radiation levels detected in the RB HVAC air exhaust or in the refueling area air exhaust automatically isolate the containment purge and vent isolation valves.

Based on the above review of the information in the DCD, the staff finds that the ESBWR CIS design meets the requirements of post-TMI Generic Issue Item II.E.4.2, "Containment Isolation Dependability" as per 10 CFR 50.34(f)(2)(xiv) and follows the guidance provided in SRP Section 6.2.4 and therefore, is acceptable.

II.E.4.4, Containment Purging During Reactor Operation (10 CFR 50.34(f)(2)(xv)

The governing regulation for TMI Action Plan Item II.E.4.4, Containment Purging During Reactor Operation, 10 CFR 50.34(f)(2)(xv), states :

> Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

The DCD entry for this generic issue, in Tier 2, Table 1A-I, simply asserts that the ESBWR design complies with these requirements, without explanation or justification.

The first requirement of the regulation refers to a situation that generally does not occur in a plant with an inerted containment atmosphere, which is unwarranted or excessive containment purging. The NRC established this generic issue because it had found that some (noninerted) plants were purging/venting their containments for sizable fractions of the plant's operating time, or even continuously. The NRC recognized that an open purge/vent line constitutes a sizable hole in the containment boundary, which is intrinsically a less safe condition than having all purge/vent valves closed, in case an accident occurs.

One legitimate reason for purging while the reactor is operating is to reduce the concentration of airborne radioactive material in the containment atmosphere, which would reduce the occupational exposure of personnel who enter containment. The regulation, then, calls for minimized purging time, consistent with as low as reasonably achievable (ALARA) principles for occupational exposure. However, personnel do not enter containments while they are inerted, so there is no need to purge for this reason. In general, plants with inerted containment will naturally minimize purge/vent time (except when inerting or de-inerting) because of the cost of the nitrogen gas needed to replace that which is expelled from containment. Also, as mentioned before, personnel exposure during containment entries is not a factor. Despite these facts, the applicant must provide a discussion in the DCD that presents these or similar arguments to demonstrate compliance with the requirement of 10 CFR 50.34(f)(2)(xv).

The second requirement of the regulation (i.e., to provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions) is explained in more detail in NUREG–0737, Item II.E.4.2, subpart (6) and Attachment 1. The staff had found that some purge/vent valves (typically butterfly valves) in operating plants were not capable of closing if a design-basis LOCA occurred while the valves were open.

In a design-basis LOCA, containment pressure increases so rapidly that the containment atmosphere rushes out through open purge/vent valves before they can begin to close. Some valves were found to be incapable of closing against the aerodynamic forces induced by the rapidly moving gas; in fact, some valves would even be damaged by the transient so that they would be stuck open and incapable of closing again until repaired. The regulation, therefore, requires the applicant to demonstrate, by analysis and/or testing, that the purge/vent valves would be capable of closing under these conditions. An alternative to such demonstration is to ensure that purge/vent valves will never be open while the plant is operating, by including a requirement in the TS that the valves must be locked or sealed closed in Modes 1 through 4, with no exception for even momentary opening of a purge/vent line while in Modes 1 through 4.

In RAI 6.2-179, the staff requested that the applicant provide the following information in the DCD to demonstrate compliance with the requirements of 10 CFR 50.34(f)(2)(xv):

- Containment purging/venting capability is designed to minimize the purging time consistent with ALARA principles for occupational exposure.

- There is high assurance that the purge system will reliably isolate under accident conditions, or the applicant should provide TS which require purge/vent valves to be sealed closed in Modes 1 through 4.

- The applicant should identify all purge/vent valves. This includes all CIVs in lines that perform a purging or venting function.

RAI 6.2-179 was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Chapter 16, TS SR 3.6.1.3.1, to eliminate the specific sizes of the purge/vent valves, and DCD Tier 2, Chapter 16B, TS SR 3.6.1.3.1, “Bases,” to include the 25-mm (1 in.), 350-mm (13.8 in.), and 400-mm (15.7 in.) purge/vent valves, as well as the 500-mm (19.7 in.) purge/vent valves. These other purge/vent valves exist within the same system (described below) as the 500-mm (19.7 in.) valves. The other systems that penetrate containment and have direct contact with the containment atmosphere (the process radiation monitoring system and the containment monitoring system) do not have a purge/vent capability. GEH provided the following information in response to RAI 6.2-179:

- The containment purging/venting is performed using the containment inerting system. DCD Tier 2, Section6.2.5.2A, describes this system. The containment inerting system is used to establish and maintain an inert atmosphere within the containment during all plant operating modes, except during plant shutdown for refueling or maintenance and during limited periods of time to permit access for inspection and maintenance during reactor low-power operation. The system is designed to permit de-inerting the containment for safe operator access and minimizing personnel exposure. DCD Tier 2, Chapter 16, TS SR 3.6.1.8, sets out the conditions for inerting and de-inerting containment (see the response to RAI 16.2-110, Supplement 2 in MFN 07-025, Supplement 2). The applicant revised DCD Section 6.2.5.1.1 to describe the function of the containment inerting system in relation to minimizing personnel exposure.

- As discussed in DCD Tier 2, Section 3.9.3.5, valves that perform an active safety-related function will be functionally qualified to perform their required functions, using ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as guidance. A qualification specification (i.e., purchase specification), consistent with Appendices QV-I and QV-A to QME-1, will be prepared for the containment purge valves to ensure that the operating conditions and safety functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility. In addition, as discussed in the DCD markup of Tier 2, Revision 4, Section 3.9.6.8 (MFN 08-131), active safety-related valves, including the containment purge valves, will be pre-operationally tested to verify that they are properly set to perform their required functions. Finally, the containment purge valves will be periodically tested as shown in DCD Tier 2, Revision 4, Table 3.9-8, as part of the inservice testing program. This testing includes periodic valve exercise testing (including stroke time measurement), verification of fail-safe performance, local leakage rate testing, and remote position indicator tests.

- Containment purging/venting is performed using the containment inerting system. A complete list of CIVs for this system appears in DCD Table 6.2-36, 6.2-37, and 6.2-38.

The applicant revised the DCD to show the specific design information of purge valves, which the staff finds acceptable. Based on the above review, the staff finds the GEH response to RAI 6.2-179 demonstrates compliance with the requirements of 10 CFR 50.34(f)(2)(xv) and follows guidance provided in SRP Section 6.2.4 and therefore, is acceptable. Based on the applicant response, RAI 6.2-179 is resolved, and the CIS meets the requirements of post-TMI Generic Issue Item II.E.4.4.

6.2.4.4 *Conclusions*

On the basis of its review, the staff concludes that the proposed ESBWR CIS, described in the DCD, complies with the acceptance criteria of Section 6.2.4 of the SRP. Compliance with the criteria in Section 6.2.4 of the SRP, as described in this section, constitutes an acceptable basis for satisfying the CIS requirements of GDC 1, 2, 4, 16, 54, 55, 56, and 57 and the additional TMI-related requirements of 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xv).

6.2.5 Combustible Gas Control in Containment

During certain accidents, combustible gases could be generated inside containment and, if not controlled, might burn and threaten the operability of the containment or various systems inside the containment that are important to safety.

6.2.5.1 *Regulatory Criteria*

The requirements for the control of combustible gas in containment during accidents appear in 10 CFR 50.44. The NRC extensively revised 10 CFR 50.44 in 2003, made associated changes to 10 CFR 50.34 and 10 CFR 52.47, and added a new section, 10 CFR 50.46a. The revisions consolidate combustible gas control regulations for future power reactor applicants and licensees and also apply to current power reactor licensees. The purpose of the revisions was to risk-inform the requirements for combustible gas control. The revised rules eliminate the former requirements for hydrogen recombiners and hydrogen purge systems and relax the former requirements for hydrogen- and oxygen-monitoring equipment to make them commensurate with their risk significance.

For the design certification of the ESBWR design, 10 CFR 50.44 requires the following:

- 10 CFR 50.44(c)(2): The containment must either (1) have an inerted atmosphere, or (2) limit hydrogen concentrations in containment during and following an accident that releases an amount of combustible gas equivalent to that generated by a 100-percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident-mitigating features. In 10 CFR 50.44(a)(1) "inerted atmosphere" is defined as "a containment atmosphere with less than 4 percent oxygen by volume."

- 10 CFR 50.44(c)(1): The containment must be capable of ensuring a mixed atmosphere during DBAs and significant beyond design basis accidents (BDBAs). The rule states that "mixed atmosphere" means that "the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity."

- 10 CFR 50.44(c)(4)(i): Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant BDBA for combustible gas control and accident management, including emergency planning.

- 10 CFR 50.44(c)(4)(ii): Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant BDBA for accident management, including emergency planning.

- 10 CFR 50.44(c)(5): The applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from a 100-percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

The appropriate staff guidance documents for this review are RG 1.7 and SRP Section 6.2.5. The staff is using Revision 3 of both documents, even though they were not formally issued until March 2007, which was after the ESBWR DCD was docketed. These revisions were issued to support the 2003 revision to 10 CFR 50.44. Draft versions of the guidance documents have been publicly available since 2003 and were substantially like the final versions. The applicant has cited RG 1.7, Revision 3 in the DCD.

The following regulations also have a bearing on this review:

- GDC 5, "Sharing of structures, systems, and components," as it relates to providing assurance that sharing of SSCs important to safety among nuclear power units will not significantly impair their ability to perform their safety functions

- GDC 41 as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; systems being designed to suitable requirements (i.e., that suitable redundancy in components and features exists) and suitable interconnections to ensure that, for either a loss of onsite or offsite power, the system safety function can be accomplished, assuming a single failure; and systems being provided with suitable leak detection, isolation, and containment capability to ensure that system safety function can be accomplished

- GDC 42, "Inspection of containment atmosphere cleanup systems," as it relates to the design of the systems to permit appropriate periodic inspection of components to ensure the integrity and capability of the systems

- GDC 43, "Testing of containment atmosphere cleanup systems," as it relates to the systems being designed to permit periodic testing to ensure system integrity and the operability of the systems and active components

- 10 CFR 52.47(b)(1), which requires that a design certification application 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 met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the Commission's rules and regulations

6.2.5.2 *Summary of Technical Information*

The design of the ESBWR provides for an inerted containment (with oxygen concentration in the containment maintained at less than 4 percent by volume) during normal operation, according to 10 CFR 50.44(c)(2), and as a result, no system to limit hydrogen concentration is required.

DCD Tier 2, Revision 9, states that the ESBWR meets the relevant requirements of the following:

- 10 CFR 50.44 and 10 CFR 50.46, as they relate to BWR plants being designed to have containments with an inerted atmosphere.

- GDC 5 does not apply to the inerting function because there is no sharing of SSCs between different units.

- GDC 41, as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained, does not apply to the ESBWR because the safety function is accomplished by keeping the containment inerted. Thus, no redundancy or single-failure criteria shall be considered, as the inerted containment is intrinsically safe and passive

- GDC 42 and GDC 43, as they relate to the design of the systems to permit appropriate periodic inspection and periodic testing of components to ensure the integrity and capability of the systems, do not apply to the inerting function. Periodic monitoring of oxygen concentration is adequate to confirm the safety function.

- RG 1.7, Revision 3, as it relates to the systems being designed to limit the oxygen gas concentrations within the containment.

Containment Inerting System: The containment inerting system is provided to establish and maintain an inert atmosphere within the containment (oxygen concentration below the maximum permission limit of 4 percent during normal power operation) as discussed in DCD Tier 2, Revision 9, Section 6.2.5.2. The containment inerting system can be used under postaccident conditions for containment atmosphere dilution to maintain an inerted condition by a controlled purge of the containment atmosphere with nitrogen to prevent reaching a combustible gas condition.

Containment Atmosphere Monitoring: The containment monitoring system discussed in DCD Tier 2, Revision 9, Section 6.2.5.3, provides the function that is necessary to meet or exceed the requirements of 10 CFR 50.44(c)(4) with regard to oxygen and hydrogen monitoring. The containment monitoring system is a safety-related, seismic Category 1 system consisting of two redundant, physically and electrically independent postaccident monitoring divisions. Each division is capable of measuring and recording the radiation levels and the oxygen and hydrogen concentration levels in the drywell and suppression chamber.

Hydrogen and Oxygen Monitoring: This system, discussed in DCD Tier 2, Revision 9, Sections 6.2.5.3.1 and 6.2.5.3.2, respectively, consists of two hydrogen- and two oxygen-monitoring channels containing hydrogen and oxygen sensors, sample lines to bring a sample from the drywell or suppression chamber to the sensor, hydrogen and oxygen monitor electronics assemblies, visual displays, and a calibration gas supply. The data are transmitted to the MCR where they are continuously displayed. High hydrogen and oxygen concentration alarms are provided. The channels are equipped with an inoperative alarm to indicate malfunctions. The channels are divided into two redundant divisions.

Radiation Monitoring: This system, discussed in DCD Tier 2, Revision 9, Section 6.2.5.3.3, consists of two channels per division (1 and 2) of radiation detector assemblies, radiation electronic assemblies and visual displays. The channels measure gross gamma radiation in the drywell and suppression chamber. The signals are carried back to the MCR where the signals are continuously displayed. The channels are equipped with an alarm to indicate channel malfunction. The radiation monitoring channels are divided into two redundant measurement divisions.

Containment Atmosphere Mixing: The ESBWR design provides protection from localized combustible gas deflagrations, including the capability to mix the steam and noncondensable gases throughout the containment atmosphere and minimize the accumulation of high concentrations of combustible gases in local areas. DCD Tier 2, Revision 9, Section 6.2.5.3.4, discusses in detail how adequate mixing within the ESBWR containment system is assured based on the configuration of the containment, coupled with the dynamics of the design-basis LOCA and the mitigating components within the containment volume.

Containment Overpressure Protection: The pressure capability of the ESBWR containment vessel is such that it will not be exceeded by any design-basis or special event. The pressure capability of the containment's limiting component is greater than the pressure that results from assuming a 100-percent fuel clad-coolant reaction. There is sufficient margin to the containment pressure capability such that there is no need for an automatic containment overpressure protection system. In a hypothetical situation in which containment depressurization is required, manual operator action can perform this depressurization.

Containment Structural Integrity: DCD Tier 2, Appendix 19B presents the deterministic analysis performed and results obtained for the containment ultimate capability under internal pressure in accordance with the requirements in 10 CFR 50.44(c)(5) and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993. Section 19.2 of this report presents the evaluation of containment structural integrity.

Postaccident Radiolytic Oxygen Generation: For a design-basis LOCA in the ESBWR, the ADS would depressurize the reactor vessel and the GDCS would provide gravity-driven flow into the vessel for emergency core cooling. The safety analyses show that the core does not uncover during this event and, as a result, there is no fuel damage or fuel clad-coolant interaction that would result in the release of fission products or hydrogen. Thus, for a design-basis LOCA, the generation of postaccident oxygen would not result in a combustible gas condition, and a design-basis LOCA does not have to be considered in this regard.

For the purposes of postaccident radiolytic oxygen generation for the ESBWR, a severe accident with a significant release of iodine and hydrogen is more appropriate to consider. Because the ESBWR containment is inerted, the prevention of a combustible gas deflagration is assured in the short term following a severe accident. In the longer term, an increase in the oxygen concentration would result from the continued radiolytic decomposition of the water in the containment. Because the possibility of a combustible gas condition is oxygen limited for an inerted containment, it is important to evaluate the containment oxygen concentration versus time following a severe accident to ensure that sufficient time will be available to implement severe accident management (SAM) actions. It is desirable to have at least a 24-hour period following an accident to allow for SAM implementation.

The DCD states that the radiolytic oxygen concentration in containment was analyzed consistent with the methodology of Appendix A to SRP Section 6.2.5 and RG 1.7.

The analysis results show that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100 percent. The results support the conclusion that sufficient time will be available to activate the emergency response organization and implement the SAM actions necessary to preclude a combustible gas deflagration.

6.2.5.3 *Staff Evaluation*

6.2.5.3.1 Combustible Gas Control

The ESBWR design specifies that the containment will be inerted with nitrogen gas during normal operation. This means that the concentration of oxygen in the containment atmosphere will be maintained at less than 4 percent by volume while the reactor is in operation. This satisfies the requirement of 10 CFR 50.44(c)(2) and is therefore acceptable.

There was, however, an open item concerning the placement of a 4 percent by volume limitation on containment oxygen concentration in the TS. In RAI 16.2-110, the staff asked GEH to propose TS Section 3.6, "Containment Systems," for containment oxygen concentration. GEH asserted that an operating restriction on oxygen concentration (to less than 4 percent by volume) is not required as an initial condition in the analysis of any design-basis event, so it does not meet Criterion 2 of 10 CFR 50.36, and thus it is not included in the proposed TS.

However, both the staff and the nuclear industry's Technical Specification Task Force have stated in the following that such TS are required:

- When the NRC revised 10 CFR 50.44 in 2003, the staff issued a model safety evaluation for implementation of the revised rule through the Consolidated Line Item Improvement Process, The model safety evaluation states, on page 13, that "…requirements for primary containment oxygen concentration will be retained in TS for plant designs with an inerted containment." Furthermore, the current standard TS for BWR/4 plants (NUREG–1433, Rev. 3 Vol. 1,) include TS 3.6.3.2, Primary Containment Oxygen Concentration, which states that "The primary containment oxygen concentration shall be < 4.0 volume percent."

- Technical Specification Task Force Traveler (TSTF)-447, Revision 1, "Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors," dated July 18, 2003, which the staff has accepted, states, "For plant designs with an inerted containment, the requirement for primary containment oxygen concentration will be retained in Technical Specifications."

In light of these positions, the staff requested that GEH add a TS limiting containment oxygen concentration to less than 4 percent by volume. RAIs16.2-110 and S01-S02 were being tracked as open items in the SER with open items.

GEH responded to RAI 16.2-110 and RAIs 16.2-110 S01-S02 by agreeing to provide a new TS 3.6.1.8, "Containment Oxygen Concentration," and associated bases in DCD Chapters 16 and 16B, respectively. In addition, GEH deleted Availability Control (AC) 3.6.1, "Containment Oxygen," from DCD Tier 2, Chapter 19.

GEH also proposed to incorporate a new special operation TS. TS 3.10.9, "Oxygen Concentration—Startup Test Program," will allow suspension of requirements of LCO 3.6.1.8 for the first 120 effective-full-power days, during performance of startup tests.

To allow containment entry for required startup tests without increasing personnel risks due to the oxygen-deficient atmosphere, GEH stated that the proposed TS 3.10.9 is generally consistent with NUREG–0123, "Standard Technical Specifications for General Electric Boiling Water Reactors," BWR/4 standard technical specifications, and TS 3.10.5, "Oxygen Concentration," as modified and presented in NEDC-31681, "BWR Owner's Group Improved

Technical Specification," for BWR/4 improved TS 3.10.12, "Oxygen Concentration—Startup Test Program."

The staff finds the GEH response adds a TS limiting containment oxygen concentration to less than 4 percent by volume as requested. Based on the applicant's response, RAI 16.2-110 and RAIs 16.2-110 S01-S02 are resolved.

6.2.5.3.2 Mixed Atmosphere

The staff reviewed the capability of the ESBWR design to ensure a mixed atmosphere during DBAs and significant BDBAs.

In RAI 6.2-138, the staff requested that GEH provide additional description of the design's capability to ensure a mixed containment atmosphere. GEH was asked to address the following: passive features of the design, including containment/subcompartment layout, elevations, and openings between compartments that impact mixing; active features of the design, including ventilation systems, cooling systems, and spray systems; and the effectiveness of the passive and active features in providing a mixed atmosphere in the design-basis and significant beyond-design-basis events. If nonsafety-related systems are relied on for mixing, the availability of these systems in the frequency dominant beyond-design-basis events and any "special treatment" requirements for these systems should also be addressed. RAI 6.2-138 was being tracked as an open item in the SER with open items.

In response to RAI 6.2-138, GEH revised or proposed changes to DCD Tier 2, Section 6.2.5. The following is the staff's evaluation of the containment mixing portion of the DCD.

The drywell and wetwell are inerted with nitrogen to meet 10 CFR 50.44. Containment mixing is not as critical for inerted containments as it is for plants with mitigative features that recombine hydrogen and oxygen. In DCD Tier 2, Section 6.2.5.3.4, GEH described the design features to ensure sufficient mixing for the drywell, wetwell, drywell head region, and RSA. The staff acknowledges that these features ensure that postaccident steam and entrained noncondensable gases will be transported to the PCCS heat exchangers. The PCCS heat exchangers are designed to condense the steam and transfer the majority of the noncondensable gases to the wetwell by the PCCS heat exchanger vent line. Another consideration with respect to the mixing process is the incorporation of passive autocatalytic recombiners (PAR)s into both the drywell and wetwell. They have been included to assist in long-term pressure control and as defense-in-depth protection against the potential buildup of combustible gases generated by the radiolytic decomposition of water. DCD Tier 2, Revision 9, Section 6.2.5.1, describes the PARs.

PARs are passive devices that operate when the surrounding atmosphere contains a stoichiometric mix of hydrogen and oxygen. The PARs contain a catalyst that facilitates the recombination of hydrogen and oxygen gases into water vapor. They also create convective air currents (recombination is an exothermic reaction), which further the recombination process and mixing within the drywell and wetwell atmosphere.

The number and size of PARs to be used in each containment compartment will be selected based on the nominal hydrogen depletion rate of each individual PAR unit such that the total depletion rate is twice the maximum hydrogen generation rate at 72 hours. The maximum hydrogen generation rate at 72 hours is 0.32 kilograms per hour (0.71 pounds per hour), based on the methodology of RG 1.7 and the analytical assumptions in DCD Tier 2, Section 6.2.5.5.2.

The number and size of PARs specified will provide the minimum safety factor of 2 for each containment compartment (drywell and wetwell) to account for possible catalytic poisons.

The minimum capacity will be the equivalent of one full-size PAR unit specified for each containment compartment; however, because of other design considerations, more and smaller capacity units (with equivalent total capacity) will be specified. This will result in more complete coverage of the wetwell and drywell. The nominal hydrogen depletion rates for the full-size PAR will be a minimum of 0.8 kilograms per hour (1.8 pounds per hour). The PARs are sited with consideration of factors such as protection from jet impingement, protection from containment spray and cooling fan discharge, protection from flooding and PS, discharged exhaust impacts, and accessibility for testing.

The staff reviewed the information provided by GEH in response to RAI 6.2-138 and finds it acceptable because the applicant revised the DCD to provide specific design criteria for the PARs consistent with RG 1.7. The ESBWR design meets 10 CFR 50.44(c)(1), based on DCD Tier 2, Revision 6, Sections 6.2.5.1 and 6.2.5.3.4 4 because of passive features of the containment design for ensuring a mixed atmosphere during design-basis and significant beyond design basis accidents. Based on the applicant's response, RAI 6.2-138 is resolved.

6.2.5.3.3 Oxygen Monitor

The regulation in 10 CFR 50.44(c)(4)(i) requires that equipment for monitoring oxygen be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant BDBA for combustible gas control and accident management, including emergency planning.

In RAI 6.2-137, RAI 6.2-137 S01 and RAI 6.2-137 S02, the staff requested additional information concerning the range of measurement of the oxygen monitors and their functionality, reliability, and accuracy, and justification that the proposed monitors are adequate for their intended function. The RAI also inquired about functionality and reliability of the monitors when exposed internally to the temperature, pressure, humidity, and radioactivity of the containment atmosphere during a significant BDBA. RAI 6.2-137, RAI 6.2-137 S01 and RAI 6.2-137 S02 were being tracked as open items in the SER with open items.

In response to RAI 6.2-137, GEH stated that equipment chosen for oxygen monitoring will be specified to meet the environmental and radiological requirements for its location and for intended postaccident operations. GEH also stated that internal components will be evaluated to ensure that the instrument is qualified for the intended environmental and radiological conditions expected and for the required postaccident monitoring timeframe.

With respect to the accuracy of the oxygen monitors, GEH responded that it would comply with Table 2 in RG 1.97, Revision 3, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," issued May 1983, where the required instrument range for a Type C variable is 0 to 10 percent volume for oxygen in inerted containments.

GEH stated that it would revise DCD Tier 2, Section 7.5.2.1, and add a new table, Table 7.5-5 (the markup was provided in the GEH response) indicating the required instrument range. The staff confirmed that the applicant incorporated this change in DCD Tier 2, Revision 6.

As described in DCD Tier 2, Revision 9, Sections 6.2.5 and 7.5.2, the oxygen monitors are a safety-related, seismic Category 1 system consisting of two redundant, physically and electrically independent postaccident monitoring divisions. The oxygen monitors are environmentally qualified (EQ). DCD Tier 2, Revision 9, Section 19.3.4.2 identifies the oxygen monitors as equipment required for severe accident mitigation. Section 19.3.3.3.8 of this report evaluates the survivability of the oxygen monitors. The oxygen monitors are located outside the drywell and wetwell, as shown in DCD Tier 2, Revision 9, Figure 7.5-1.

The staff reviewed the information provided by GEH for oxygen monitor in response to RAI 6.2-137, RAI 6.2-137 S01 and RAI 6.2-137 S02 and finds them acceptable because the applicant revised the DCD to provide specific design criteria for the oxygen monitor consistent with RG 1.7 and RG 1.97. The ESBWR design of the oxygen monitors meets the regulation of 10 CFR 50.44(c)(4)(i) in accordance with the guidelines of SRP Section 6.2.5 and therefore is acceptable. Based on the applicant's responses, RAI 6.2-137, RAI 6.2-137 S01 and RAI 6.2-137 S02 are resolved.

6.2.5.3.4 Hydrogen Monitor

The regulation in 10 CFR 50.44(c)(4)(I) requires that equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant BDBA for combustible gas control and accident management, including emergency planning.

In RAI 6.2-136, RAI 6.2-136 S01 and RAI 6.2-136 S02, the staff requested that GEH provide additional information concerning the range of measurement of the hydrogen monitors and their functionality, reliability, and accuracy, and justification that the proposed monitors are adequate for their intended function. The RAI also inquired about the functionality and reliability of the monitors when exposed internally to the temperature, pressure, humidity, and radioactivity of the containment atmosphere during a significant BDBA. RAI 6.2-136 and its supplements were being tracked as open items in the SER with open items.

In response to RAI 6.2-136, GEH stated that equipment chosen for hydrogen monitoring will be specified to meet the environmental and radiological requirements for its location and for intended postaccident operations. GEH also stated that internal components will be evaluated to ensure that the instrument is qualified for the intended environmental and radiological conditions expected and for the required postaccident monitoring timeframe.

With respect to the accuracy of the hydrogen monitors, GEH responded that it will comply with RG 1.97, Revision 3, Table 2, where the required instrument range for a Type C variable is 0 to 30-percent volume for hydrogen in inerted containments.

GEH stated that it would revise DCD Tier 2, Section 7.5.2.1, and add a new table, Table 7.5-5(a) markup table was provided in the GEH response) indicating the required instrument range. The staff confirmed that the applicant incorporated this change in DCD Tier 2, Revision 6.

As described in DCD Tier 2, Sections 6.2.5 and 7.5.2, the hydrogen monitors are a safety-related, seismic Category 1 system consisting of two redundant, physically and electrically independent postaccident monitoring divisions. The hydrogen monitors are EQ.

DCD Tier 2, Section 19.3.4.2, identifies the hydrogen monitors as equipment required for severe accident mitigation. Section 19.2.3.3.8 of this report evaluates the survivability of the hydrogen monitors. The hydrogen monitors are located outside the drywell and wetwell, as shown in ESBWR DCD Tier 2, Revision 9, Figure 7.5-1.

The staff reviewed the information provided by GEH for hydrogen monitor in response to RAI 6.2-136, RAI 6.2-136 S01 and RAI 6.2-136 S02 and finds them acceptable because the applicant revised the DCD to provide the specific design criteria for the hydrogen monitor consistent with RG 1.7 and RG 1.97. The ESBWR design for the hydrogen monitors meets the regulation in 10 CFR 50.44(c)(4)(ii) in accordance with guidelines of SRP Section 6.2.5 and therefore is acceptable. Based on the applicant's response, RAI 6.2-136 and its supplements are resolved.

6.2.5.3.5 Structural Analysis

As required by 10 CFR 50.44(c)(5), the applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and includes sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from a 100-percent fuel clad reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

In RAI 6.2-178, the staff requested that GEH identify the design-basis and special events that were considered in the analysis and provide the actual pressure that results from assuming a 100-percent fuel clad-coolant reaction, and whether this assumption includes hydrogen burning. If no hydrogen burning was assumed for any accident, GEH should justify this omission, with consideration of BDBA information from DCD Tier 2, Chapter 19. RAI 6.2-178 was being tracked as an open item in the SER with open items.

In response, GEH stated that the design-basis and special events were those described in DCD Tier 2, Chapters 6 and 15. The estimate of the internal pressure that results from assuming a 100-percent fuel clad-coolant reaction is 1.097 MPa (absolute) (159.1 psia) as described in the response to RAI 19.2-39 in DCD Tier 2, Section 19B. The analysis did not consider burning of hydrogen because the containment is inerted.

RG 1.7, Revision 3, Section C.5, describes an analytical technique that is accepted by the staff. The applicant has used this technique in DCD Tier 2, Section 19B, and concluded that the deterministic FE analysis demonstrates that the reinforced concrete containment vessel and liner maintain structural integrity according to the requirements of 10 CFR 50.44(c)(5) for pressures corresponding to 100-percent fuel clad-coolant reaction. Section 19.2 of this report presents the evaluation of DCD Tier 2, Section 19B. The staff acknowledges that hydrogen burning was not considered because the containment is inert and analyses provided in DCD Tier 2, Section 6.2.5.5.3, show that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100 percent. The staff finds that the requirements of 10 CFR 50.44(c)(5) are met based on its evaluation in Section 19.2.4 of this report. Based on the applicant's response, RAI 6.2-178 is resolved.

6.2.5.3.6 Other Regulations

This section addresses regulations, other than 10 CFR 50.44, that relate to combustible gas control in containment. Section 6.2.5.1 of this report lists these regulations.

The ESBWR design meets the relevant requirements of the following:

- GDC 5 does not apply because there is no sharing of SSCs between different units.

- GDC 41, as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained, is satisfied because the safety function is accomplished by keeping the containment inerted. Thus, no redundancy or single-failure criteria need be considered, as the inerted containment is intrinsically passive.

- GDC 42 and GDC 43 related to the design of the systems to permit appropriate periodic inspection and periodic testing of components to ensure the integrity and capability of the systems, do not apply to the inerting function; periodic monitoring of oxygen concentration is adequate to confirm the safety function.

- 10 CFR 52.47(b)(1) relates to ITAAC. Section 14.3.11 of this report addresses ITAAC related to containment and associated systems.

6.2.5.4 *Conclusions*

On the basis of its review, the staff concludes that the proposed ESBWR combustible gas control system in the containment, described in the DCD, complies with the acceptance criteria of Section 6.2.5 of the SRP.

Compliance with the criteria in Section 6.2.5 of the SRP, as described in this section, constitutes an acceptable basis for satisfying the combustible gas control system requirements in 10 CFR 50.44, GDC 41, 42, and 43, and the guidance in RG 1.7.

6.2.6 Containment Leakage Testing

DCD Tier 2, Revision 9, Section 6.2.6, describes the proposed containment leakage rate testing program for the ESBWR.

6.2.6.1 *Regulatory Criteria*

Conformance with the requirements of either Option A or B of Appendix J to 10 CFR Part 50, and the provisions of RG 1.163, "Performance-Based Containment Leak-Test Program," September 1995, constitutes an acceptable basis for satisfying the requirements of the following GDC applicable to containment leakage rate testing, in accordance with SRP Section 6.2.6, Revision 3:

- GDC 52, “Capability for containment leakage rate testing,” as it relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the containment integrated leakage rate test (up to the containment design pressure)

- GDC 53, as it relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leakage rate testing, at the containment design pressure, of penetrations having resilient seals and expansion bellows

- GDC 54, as it relates to piping systems that penetrate primary reactor containment being designed with a capability to determine whether the valve leakage rate is within acceptable limits

- 10 CFR 52.47(b)(1), which requires that a design certification application 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 met, a plant that incorporates the design certification is built and will operate in accordance with the design certification; the provisions of the Atomic Energy Act of 1954, as amended; and the NRC's regulations

6.2.6.2 *Summary of Technical Information*

This section describes the testing program for determining the containment integrated leakage rate (Type A tests), containment penetration leakage rates (Type B tests), and CIV leakage rates (Type C tests) that complies with Option A or B of Appendix J to 10 CFR Part 50, in accordance with RG 1.163 and GDC 52, 53, and 54. The leakage rate testing capability is consistent with the testing requirements of ANSI/ANS 56.8-2002, "Containment System Leakage Testing Requirements," dated November 2002.

Licensees perform Type A, B, and C tests before operation and periodically thereafter to ensure that leakage rates through the containment and through systems or components that penetrate containment do not exceed the maximum allowable rates. Containment maintenance, including repairs of systems and components penetrating the containment, is performed as necessary to maintain leakage rates at or below acceptable values.

6.2.6.2.1 Containment Integrated Leakage Rate Test (Type A)

ILRTs (Type A tests) are conducted periodically, in conformance with Appendix J to 10 CFR Part 50, to ensure that containment integrity is maintained and to determine whether the leakage rate has increased since the previous ILRT. The tests are performed after major repairs and upon indication of excessive leakage. Verification tests are also performed after each ILRT. After the initial ILRT, periodic ILRTs will be performed at intervals, depending on whether the COL licensee selects Option A or Option B of Appendix J to 10 CFR Part 50. If the COL licensee selects Option A, it will perform the ILRTs at least three times during each 10-year service period. If it selects Option B, the test interval will be in accordance with RG 1.163.

In addition, after the initial ILRT, the COL licensee will follow any major modification or replacement of components of the reactor containment with either a Type A or a Type B test, ensuring that the area affected by the modification meets the applicable acceptance criteria.

A standard statistical analysis of the data is conducted by a linear regression analysis, using the method of least squares to determine the leakage rate and associated 95-percent upper confidence limit (UCL). ILRT results are satisfactory if the UCL is less than 75 percent of the maximum allowable leakage rate, L_a. As an exemption from the definition of L_a in Appendix J to 10 CFR Part 50, L_a is redefined as "containment leakage rate" in DCD Tier 2, Revision 9, Table 6.2-1, which excludes the MSIV leakage rate. The treatment of an MSIV leakage pathway separately in the radiological dose analysis in DCD Tier 2, Revision 9, Section 15.4.4.5.2, justifies this exemption.

After completing the initial ILRT, a verification test is conducted to confirm the ability of the ILRT method and equipment to satisfactorily determine the containment leakage rate. The accuracy of the leakage rate tests is verified by superimposing a calibrated leak on the normal containment leakage rate or by other methods of demonstrated equivalency. The difference between the total leakage and the superimposed known leakage is the actual leakage rate.

This method confirms the test's accuracy. The measurements are acceptable if the correlation between the verification test data and the ILRT data demonstrates an agreement within ± 0.25 L_a. Appendix C to ANSI/ANS-56.8 includes more descriptive information on verification methods.

During the ILRT (including the verification test), if excessive leakage occurs through locally testable penetrations or isolation valves, to the extent that it would interfere with the satisfactory completion of the test, these leakage paths may be isolated and the Type A test continued until completion. A local test shall be performed before and after the repair of each isolated path. The test results shall be reported with both pre-and post-repair local leakage rates, as if two Type A tests had been conducted. A record of corrective actions shall be documented as described below:

- For Option A of Appendix J to 10 CFR Part 50, the sum of the local leakage rates and the UCL shall be less than 0.75 L_a. Local leakage rates shall not be subtracted from the Type A test results to determine the acceptability of the test.

- For Option B of Appendix J to 10 CFR Part 50, the acceptance criteria shall be based on a calculated performance leakage rate that is defined as the sum of the Type A UCL and the as-left minimum pathway leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position (i.e., drained and vented to the containment atmosphere) before performing the Type A test. In addition, any leakage pathways that were isolated during the test shall be factored into the performance determination. If the leakage can be determined by a local leak-rate test, the as-left minimum pathway leakage rate for that leakage path must also be added to the Type A UCL. If the leakage cannot be determined by local leak-rate testing, the performance criteria for the Type A test are not met.

If the COL licensee selects Option A of Appendix J to 10 CFR Part 50, and if two consecutive periodic ILRTs fail to meet the acceptance criteria before corrective action, the COL licensee will perform an ILRT at each plant shutdown for major refueling or approximately every 24 months (whichever occurs first), until two consecutive ILRTs meet the acceptance criteria, after which time, the COL licensee may resume the previously established periodic retest schedule.

If the COL licensee selects Option B of Appendix J to 10 CFR Part 50, and if the ILRT results are not acceptable, then the COL licensee should identify the cause of the unacceptable performance and determine appropriate corrective actions.

Once the COL licensee has determined the cause and has completed the corrective actions, it should reestablish acceptable performance by performing an ILRT within 48 months following the unsuccessful ILRT test. Following a successful ILRT, the surveillance frequency may revert to once every 10 years.

The additional criteria below will be met for ILRTs, if the COL licensee chooses Option A of Appendix J to 10 CFR Part 50:

- The following portions of systems are kept open or vented to the containment atmosphere during the ILRT:

- Portions of fluid systems that are part of the RCPB that are open directly to the reactor containment atmosphere under postaccident conditions and that become an extension of the boundary of the reactor containment
- Portions of closed systems inside containment that penetrate containment and that are not relied upon for containment isolation purposes following a LOCA
- Portions of closed systems inside containment that penetrate containment and rupture as a result of a LOCA

Note, however, that the ESBWR does not have any system that penetrates the containment and ruptures as a result of a LOCA.

- All systems not designed to remain filled with fluid (e.g., vented) after a LOCA are drained of water to the extent necessary to ensure exposure of the system CIVs to the containment air test pressure.

- Those portions of fluid systems penetrating containment that are external to the containment and that are not designed to provide a containment isolation barrier are vented to the outside atmosphere, as applicable, to ensure that the full postaccident differential pressure is maintained across the containment isolation barrier.

- Systems that are required to maintain the plant in a safe condition during the ILRT are operable in their normal mode and are not vented. Also, systems that are normally filled with water and operating under post-LOCA conditions are not vented. The results of local leakage rate tests of penetrations associated with these systems are added to the ILRT results.

The additional criteria below will be met for ILRTs if the COL licensee chooses Option B of Appendix J to 10 CFR Part 50. All Appendix J pathways must be properly drained and vented during the ILRT, with the following exceptions:

- Pathways in systems that are required for proper conduct of the ILRT or to maintain the plant in a safe-shutdown condition during the ILRT

- Pathways in systems that are normally filled with fluid and operable under postaccident conditions

- Portions of pathways outside primary containment that are designed to seismic Category I and at least Safety Class 2

- For planning and scheduling purposes, or ALARA considerations, pathways that are Type B or C tested within the previous 24 calendar months that need not be vented or drained during the ILRT

6.2.6.2.2 Containment Penetration Leakage Rate Test (Type B)

Containment penetrations designed to incorporate resilient seals, bellows, gaskets, or sealant compounds; air locks and air-lock door seals; equipment and access hatch seals; and electrical penetration canisters receive preoperational and periodic Type B leakage rate tests, in accordance with Appendix J to 10 CFR Part 50. The local leak detection tests of Type B and Type C are completed before the preoperational or periodic Type A tests.

Type B tests are performed at containment peak accident pressure, P_a, by local pressurization, using either the pressure-decay or flowmeter method. For the pressure-decay method, a test volume is pressurized with air or nitrogen to at least P_a. The rate of decay of pressure of the known test volume is monitored to calculate the leakage rate. For the flowmeter method, the required test pressure is maintained in the test volume by making up air or nitrogen, through a calibrated flowmeter. The flowmeter fluid flow rate is the leakage rate from the test volume. The plant-specific TS include the acceptance criteria for Type B tests. The combined leakage rate of all components subject to Type B and Type C tests should not exceed 60 percent of L_a.

In accordance with Appendix J to 10 CFR Part 50, Type B tests are performed at intervals that depend on whether Option A or Option B is selected on a unit-specific basis. If Option A is selected, Type B tests (except for air locks) will be performed during each reactor shutdown for major fuel reloading, or other convenient intervals, but never at intervals greater than 2 years. Under this option, air locks opened when containment integrity is required are tested in manual mode within 3 days of being opened. If the air lock is to be opened more frequently than once every 3 days, it is tested at least once every 3 days during the period of frequent openings. The acceptance criterion for an air lock is a leakage rate of less than or equal to 0.05 L_a, when tested at a pressure greater than or equal to P_a.

As an exemption from Appendix J to 10 CFR Part 50, Section III.D.2.(b)(ii) can be satisfied by testing at the end of periods when containment integrity is not required by the plant's TS, at a lower test pressure specified in the TS applied between the door seals with an acceptable maximum measured leakage rate of 0.01 L_a. Air locks are tested at initial fuel loading and at least once every 6 months thereafter. If Option B is selected, the test interval will be in accordance with RG 1.163.

Air locks that are allowed to be opened during power operation may be tested at power operation so as to avoid shutting down the reactor. Personnel air locks through the containment include provisions for testing the door seals and the overall air-lock leakage rates. Each door includes test connections that allow the annulus between the seals to be pressurized, and the pressure decay (if the pressure-decay method is used) or flow (if the flowmeter method is used) is monitored to determine the leak tight integrity of the seals.

Test connections are also provided on the outer face of each bulkhead so that the entire lock interior can be pressurized and the pressure decay or flow monitored to determine the overall lock leakage. Clamps or tie-downs are installed to keep the doors sealed during the overall lock test, because normal locking mechanisms are not designed for the full differential pressure across the door in the reverse direction.

6.2.6.2.3 Containment Isolation Valve Leakage Rate Test (Type C)

Type C tests are performed on all CIVs required to be tested by either Option A or Option B of Appendix J to 10 CFR Part 50. Type C tests (like Type B tests) are performed by local pressurization, using either the pressure decay or flowmeter method. The test pressure is applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or conservative.

Valves that are sealed with a fluid from a seal system, or valves not provided with a seal system and that may be justified to be equivalent to valves with a seal system, shall be tested in accordance with Option A or Option B of Appendix J to 10 CFR Part 50.

A valid justification for the equivalency of such valves is that they are located in lines designed to be, or remain, filled with water for at least 30 days after a LOCA. All test connections, vent lines, or drain lines consisting of double or multiple barriers (e.g., two valves in series, one valve and a cap, or one valve and a flange) that are connected between isolation valves, form a part of the containment boundary, and are 25.4 mm (1 in.) or less in size, may not be Type C-tested because they are used infrequently and because the multiple barrier configurations are maintained using an administrative control program.

Type C testing shall be performed in the correct direction of the leakage path, unless it can be demonstrated that testing in the reverse direction is equivalent or more conservative. The correct direction of the leakage path is from inside the containment to outside containment. Instrument lines that penetrate the containment conform to RG 1.11 and may not be Type C-tested. The lines that connect to the RCPB include a restricting orifice inside containment, are seismic Category I, and terminate in seismic Category I instruments. The instrument lines also include manual isolation valves and excess flow check valves or equivalent.

These valves are normally open and are considered extensions of the containment, the integrity of which is continuously demonstrated during normal operation. In addition, these lines are subject to the periodic Type A test because they are open (up to the pressure boundary instruments) during the ILRT. Leaktight integrity is also verified during functional and surveillance activities, as well as by visual observations during operator tours. The combined leakage rate of all components subject to Type B and Type C tests shall not exceed 60 percent of L_a. The plant-specific TS detail the periodic leakage rate test schedule requirements for Types A, B, and C tests. Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, with test intervals that conform to either Option A or Option B of Appendix J to 10 CFR Part 50. Each time a Type B or Type C test is completed, the overall total leakage rate for all required Type B and Type C tests is updated to reflect the most recent test results.

In addition to the periodic tests, any major modification or replacement of a component that is part of the primary reactor containment boundary performed after the preoperational leakage rate test will be followed by either a Type A, B, or C test (as applicable) for the area affected by the modification. The leakage test summary report will describe the containment inspection method, any repairs necessary to meet the acceptance criteria, and the test results. Following the drywell structural integrity test, a preoperational drywell-to-wetwell leakage rate test is performed at the peak drywell-to-wetwell differential pressure. Also, drywell-to-wetwell leakage rate tests are conducted at a reduced differential pressure corresponding approximately to the submergence of the vents. These tests are performed following the preoperational ILRT and periodically thereafter. They verify that no paths exist for gross leakage from the drywell to the wetwell air space that bypass the pressure suppression pool. The combination of the peak pressure and reduced pressure leakage tests also verifies adequate performance of the drywell over the full range of postulated primary system break sizes.

Drywell-to-wetwell leakage rate tests are performed with the drywell isolated from the wetwell. Valves and system lineups are the same as for the ILRT, except for paths that equalize drywell and wetwell pressure, which are open during the ILRT and are isolated during the drywell leakage test. The drywell atmosphere is allowed to stabilize for a period of 1 hour after attaining the test pressure. Leakage rate test calculations, using the wetwell pressure rise method, commence after the stabilization period.

The pressure rise method is based on the containment atmospheric pressure and temperature observations and the known wetwell volume. The leakage rate is calculated from the pressure and temperature data, wetwell free air volume, and elapsed time.

The plant-specific TS specify the periodic drywell-to-wetwell leakage rate test pressure, duration, frequency, and acceptance criteria.

6.2.6.3 *Staff Evaluation*

The staff reviewed the information in DCD Tier 2 for conformance with the requirements of Appendix J to 10 CFR Part 50 and GDC 52, 53, and 54. The staff used the guidance, staff positions, and acceptance criteria of SRP Section 6.2.6 and RG 1.163 in conducting its review.

Meeting the requirements of Appendix J to 10 CFR Part 50 ensures that the leaktightness of the containment will be within the values specified in the facility TS and that offsite radiation doses in excess of the reference values specified in 10 CFR Part 100, “Reactor Site Criteria,” will not occur. Chapter 14 of this report addresses both 10 CFR 52.47(b)(1), as it relates to ITAAC, and the ITAAC themselves.

Based on its review, the staff had two open items, RAI 6.2-90 and 6.2-91 S01. RAI 6.2-90 and RAI 6.2-91 S01 were being tracked as open items in the SER with open items. RAI 6.2-90 and RAI 6.2-90 S01 asked GEH to (1) clarify in the DCD that “Type C” means testing with air or nitrogen and eliminating water as an allowed Type C test medium, and (2) for Options A or B, address CIV testing, in systems such as RWCU/SDC, under the requirements for seal systems.

RAI 6.2-91 and RAI 6.2-91 S01 asked that DCD revisions better reflect the regulatory requirements, as indicated, related to seal systems.

In response, GEH stated that it will revise DCD Tier 2, Section 6.2.6.3, to delete the option for water as a Type C test medium. It will also revise this section to clarify the requirements for testing CIVs with qualified seal systems, such as in the RWCU/SDC system, and to include the referenced provisions in Nuclear Energy Institute 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” issued December 8, 2005. DCD Tier 2, Section 6.2.6.3, will be revised to include the regulatory requirements related to seal systems, as shown in the attached markup. The staff reviewed the GEH response in DCD Tier 2, Revision 6, Section 6.2.6.3, and finds it acceptable. GEH clarified in the DCD that “Type C” means testing with air or nitrogen and eliminated water as an allowed test medium and addressed CIV testing, in systems such as in RWCU/SDC, under the requirements of seal systems as indicated in RAI 6.2-90. The revised DCD also includes the regulatory requirements related to seal systems as indicated in RAI 6.2-91.

Based on the applicant’s response, RAI 6.2-90 and RAI 6.2-91 S01 are resolved.

6.2.6.4 *Generic Issues*

The staff’s review of containment leakage rate testing includes one Generic Safety Issue, Item A-23, “Containment Leak Testing” (see NUREG–0933, “A Prioritization of Generic Safety Issues,” issued September 2007).

The staff addressed Item A-23 by revising and clarifying Appendix J to 10 CFR Part 50 and issuing RG 1.163, and thus, Item A-23 requires no additional review or action relative to the ESBWR.

6.2.6.5 *Conclusions*

On the basis of its review, the staff concludes that the ESBWR DCD containment leakage rate testing program complies with the acceptance criteria of SRP Section 6.2.6, as described in this section, and thus constitutes an acceptable basis for satisfying the containment leakage rate testing requirements of GDC 52, 53, and 54, and Appendix J to 10 CFR Part 50.

6.2.7 Fracture Prevention of Containment Pressure Boundary

6.2.7.1 *Regulatory Criteria*

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 6.2.7, in accordance with SRP Section 6.2.7, Revision 1, issued March 2007.

The reactor containment system includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products. It must prevent fractures of the containment pressure boundary. The ESBWR must address the following regulations:

- GDC 1 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Section 6.1.1 addresses the applicant's discussion and the staff's evaluation.

- GDC 16 requires that the reactor containment and associated systems establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Sections 6.2.3 and 6.2.4 address the applicant's discussion and the staff's evaluation.

- GDC 51, "Fracture prevention of containment pressure boundary," requires that the reactor containment boundary be designed with sufficient margins to ensure that, under operating, maintenance, testing, and postulated accident conditions, (1) its ferritic materials behave in a non-brittle manner, and (2) the probability of a rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) flaw size.

The staff reviewed the DCD to ascertain whether the containment pressure boundary materials meet the requirements of GDC 51.

6.2.7.2 *Summary of Technical Information*

The containment vessel of the ESBWR is a reinforced concrete structure with ferritic parts, such as a liner and a removable head. The ferritic parts are made of materials that have a nil ductility transition temperature sufficiently below the minimum service temperature to ensure that, under operating, maintenance, testing, and postulated accident conditions, the ferritic materials behave in a non-brittle manner, considering the uncertainties in determining the material properties, stresses, and size of flaws. In DCD Tier 2, Revision 9, Table 6.1-1, the applicant identified the containment vessel liner materials, which are in conformance with ASME Code, Section III (CC-2520, "Fracture Toughness Requirements for Materials"). This meets the requirements of GDC 51. GDC 51 is only applicable to those parts of the containment that are to be made of ferritic materials.

6.2.7.3 *Staff Evaluation*

The staff reviewed the ESBWR measures involving fracture prevention of ferritic materials used in the containment pressure boundary, in accordance with SRP Section 6.2.7. These ferritic materials are acceptable if they meet the requirements of GDC 51, as it relates to the reactor containment pressure boundary being designed with sufficient margins to ensure that, under operating, maintenance, testing, and postulated accident conditions, the ferritic materials will behave in a non-brittle manner and the probability of a rapidly propagating fracture will be minimized.

6.2.7.4 *Conclusions*

Based on the review of the information included in the ESBWR, the staff finds that the fracture toughness of the materials used in the reactor containment pressure boundary meets the fracture toughness requirements specified in GDC 51. This satisfies the requirements of GDC 51 for fracture prevention of the containment pressure boundary.

The staff, therefore, concludes that, under operating, maintenance, testing, and postulated accident conditions, the ESBWR provides reasonable assurance that the materials used in the reactor containment pressure boundary will not undergo brittle fracture and that the probability of a rapidly propagating fracture will be minimized, thereby meeting the requirements of GDC 51.

6.3 Emergency Core Cooling Systems

6.3.1 Emergency Core Cooling Systems Design

6.3.1.1 *Regulatory Criteria*

The staff reviewed DCD Tier 2, Revision 9, Section 6.3, in accordance with SRP Section 6.3 and Section 15.6.5, Revision 3, issued 6/96. The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any generic issues (GIs), bulletins (BLs), generic letters (GLs), or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 3, Section 6.3 and Section 15.6.5, issued 6/96, is acceptable for this review.

The staff based its acceptance criteria on the following requirements:

- GDC 2, as it relates to the seismic design of SSCs where their failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function

- GDC 4, as it relates to the dynamic effects associated with flow instabilities and loads (e.g., water hammer)

- GDC 5, as it relates to nuclear power units not sharing SSCs important to safety unless the applicant can demonstrate that sharing will not impair the ability of such SSCs to perform their safety function

- GDC 17, “Electric power systems,” as it relates to the design of the ECCS having sufficient capacity and capability to ensure that the system does not exceed specified acceptable fuel design limits and the design conditions of the RCPB during anticipated operational occurrences and that the core is cooled during accident conditions

- GDC 27, “Combined reactivity control systems capability,” as it relates to the ECCS design having the capability to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the system will maintain the capability to cool the core

- GDC 35, as it relates to the provision of an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling

- GDC 36, “Inspection of emergency core cooling system,” as it relates to the appropriate periodic inspection of important components

- GDC 37, “Testing of emergency core cooling system,” as it relates to periodic pressure and functional testing

- 10 CFR 50.46, as it relates to (1) the design of the ECCS, (2) ensuring that the ECCS cooling performance is calculated in accordance with an acceptable evaluation model, and (3) demonstrating that the following five major ECCS acceptance criteria are met:

    (1) The calculated maximum fuel element cladding temperature does not exceed 1,204 degrees C (2,200 degrees F).

    (2) The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation.

    (3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

    (4) Calculated changes in core geometry are such that the core remains amenable to cooling.

    (5) After any calculated successful initial operation of the ECCS, the system maintains the calculated core temperature at an acceptably low value and removes decay heat for the extended period of time required by the long-lived radioactivity.

The staff also evaluated DCD Tier 2, Revision 9, Section 6.3, for conformance with the following sections of the TMI action plan, NUREG–0737:

- TMI Action Plan Item II.K.3.15, which involves isolation of the high-pressure coolant injection and the reactor core isolation cooling for BWR plants

- TMI Action Plan Item II.K.3.18, which is equivalent to 10 CFR 50.34(f)(1)(vii), with respect to eliminating the need for manual actuation of the BWR ADS to ensure adequate core cooling

- TMI Action Plan Item II.K.3.28, which is equivalent to 10 CFR 50.34(f)(1)(x), with respect to BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for nonsafety-related equipment or instrumentation and accounting for normal expected air (or nitrogen) leakage through valves

- TMI Action Plan Item II.K.3.45, which is equivalent to 10 CFR 50.34(f)(1)(xi), with regard to providing an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs

- TMI Action Plan Item III.D.1.1, which is equivalent to 10 CFR 50.34(f)(2)(xxvi), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside the containment that contain or may contain radioactive material following an accident

6.3.1.2 *Summary of Technical Information*

In DCD Tier 2, Revision 9, Section 6.3, GEH described the ECCS and the design criteria that satisfy the NRC regulatory requirements. Below is a brief summary of the GEH description.

Passive Core Cooling System

The passive core cooling system comprises the GDCS, the ADS, the ICS, and the SLCS. The GDCS, in conjunction with the ADS, the ICS, and the SLCS, provides emergency core cooling in case of a LOCA. When it receives an initiation signal, the ADS depressurizes the reactor vessel and the GDCS injects cooling water, in addition to that supplied by the ICS and SLCS, to maintain the peak cladding temperatures (PCT)s below the limits defined in 10 CFR 50.46.

Gravity-Driven Cooling System

The GDCS is a passive makeup water system. Water flows into the vessel by gravity from the GDCS pools. This differs from the ECCS in currently operating BWR/2-6 designs, which rely on active pumps and support systems. The GDCS injects water into the downcomer annulus region of the RPV following a LOCA and reactor vessel depressurization. It provides short-term, gravity-driven water makeup from three separate water pools located within the upper drywell at an elevation above the active core region through eight separate injection nozzles in the RPV. In the long term, most of the coolant boil-off is returned to the RPV as condensate from the ICs or the PCCS heat exchangers; however, there will be some boil-off loss of inventory to the drywell. The GDCS provides long-term, post-LOCA makeup from the suppression pool to meet long-term core decay heat boil-off requirements through four separate equalizing lines.

The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate by operator action. The GDCS consists of four identical trains independent of one another, both electrically and mechanically, with the exception of two trains sharing one of the three GDCS pools. Each GDCS injection and equalizing line consists of two normally locked-open manual valves, a check valve, and a squib-actuated valve. A confirmed low RPV water-level signal or a sustained drywell high pressure actuates the ADS to reduce RPV pressure. In the GDCS logic, short-term and long-term timers simultaneously start. After timeout and satisfying permissive conditions, squib valves actuate to provide an open flow path from the water sources (GDCS pools in the short term and suppression pool in the long term) to the vessel.

In the event of a core-melt sequence that causes failure of the lower vessel head and results in molten fuel reaching the lower drywell cavity floor, the GDCS floods the lower drywell region with water through four separate deluge lines. Logic circuits receiving input signals from an array of temperature sensors in the lower drywell actuate squib valves to initiate the water flow. Actuation occurs when the lower drywell basemat temperature exceeds 537 degrees C (1,000 degrees F). Once the squib valves are actuated, the GDCS deluge lines provide a flow path from the GDCS pool to the lower drywell cavity.

Squib Valve

The ECCS uses squib-actuated valves for injection to the RPV. Specifically, the function of the squib-actuated valve is to open upon receiving a signal and to remain in its full open position without any continuing external power source and thereby to admit reactor coolant makeup into the RPV in the event of a LOCA. The valves also function in the closed position to prevent RPV backflow and to maintain the RCPB during normal plant operation. The valves are horizontally mounted, straight-through, long-duration submersible, pyrotechnic-actuated, and non-reclosing, with metal diaphragm seals and flanged ends. The valve diaphragms form part of the reactor pressure boundary. The valves actuate when either of the two squib initiators ignite, causing the valves to open. The squib valves can be refurbished once fired. Squib-actuated valves are also used in the equalizing lines and the deluge lines. To minimize the potential for common-mode failure, different batches of pyrotechnic charges are used for the equalizing valves and the GDCS injection valves, and a different booster material is used for the deluge line squib valves.

Automatic Depressurization System

The ADS is part of the ECCS and operates to depressurize the reactor so that the low-pressure GDCS can inject makeup coolant to the reactor. The ADS is composed of 10 SRVs and 8 squib-actuated DPVs and their associated instrumentation and controls. The SRVs are mounted on top of the MSLs in the drywell and discharge through lines routed to quenchers in the suppression pool. Section 5.2.2 of this report describes the SRVs and DPVs.

The DPVs are straight-through, squib-actuated, non-reclosing valves. The valve size provides about twice the depressurization capacity of the SRV. The DPVs are designed so that there is low leakage throughout the life of the valve. Two initiators (squibs), singly or jointly, actuate a booster, which actuates the shearing plunger. Either one or both of two battery-powered, independent firing circuits initiate the squibs. The firing of one initiator booster is adequate to activate the plunger. All eight DPVs are horizontally mounted on horizontal stub tubes connected to the RPV at about the elevation of the MSLs. The DPVs discharge into the drywell airspace.

Isolation Condenser System

The ICS provides additional liquid inventory upon the opening of the condensate return valves to initiate the system. The ICS also provides initial depressurization of the reactor before ADS in the event of a loss of feedwater. (Section 5.4.6 of this report contains a detailed description of the ICS.)

Standby Liquid Control System

The SLCS provides additional liquid inventory in the event of DPV actuation. The firing of squib-actuated injection valves initiates the SLCS to accomplish this function. (Section 9.3.5 of this report contains a detailed description of the SLCS.)

Strainers

Section 6.2.1.7 of this report contains a description of the strainers.

6.3.1.3 *Staff Evaluation*

The staff's review of the ECCS uses SRP Section 6.3 as guidance. Because the ESBWR ECCS is quite different from the ECCS of the existing BWR designs, some SRP guidelines do not apply. The staff devoted the major portion of the review effort to the areas where the application is not identical to previously reviewed BWRs.

Emergency Core Cooling Systems

The ECCS is designed to provide coolant inventory to the reactor coolant system in the event of a LOCA. It has sufficient capacity to make up for the loss of coolant from a large spectrum of pipe breaks, up to and including a double-ended rupture of the largest pipe carrying water or steam connected to the RCPB, as well as spurious SRV operation. The passive ECCS is a safety-related system designed to perform the emergency core cooling function. The ECCS consists of the GDCS, the ADS, the ICS, and the SLCS.

The ECCS is passive and its subsystems or components require only a one-time alignment of valves upon actuation. Once the initial actuation alignment is made, they rely solely on natural forces, such as gravity and stored energy, to operate. Once opened, the injection valves remain open and cannot be closed or overridden by operators. The use of active equipment or supporting systems, such as pumps, alternating current (ac) power sources, component cooling water, or service water, is not required for the first 72 hours following an accident.

Unlike current operating BWR/2-6 designs, the ICS and SLCS in the ESBWR design are part of the ECCS. The ICS and SLCS provide additional liquid inventory that is credited in the ESBWR LOCA analysis. The GDCS, ADS, and SLCS are initiated on low RPV Level 1 with a timer delay. The ICS injection is initiated on RPV Level 2 with a timer delay or RPV Level 1 with no timer delay. Section 3.9.6 of this report contains the staff evaluation of the DPV, GDCS, and SLCS valve tests.

Gravity-Driven Cooling System

The GDCS is an ESF system. It is classified as safety-related and seismic Category I. The GDCS instrumentation and associated dc power supply are IEEE Class 1E. The GDCS injection squib valves are opened after a 150-second delay from the ECCS initiation start signal. This time delay allows the reactor to depressurize so that the GDCS can inject into the RPV. In addition, suction from the suppression pool is initiated when the RPV level drops to Level 0.5 setpoint (1.0 m [3.28 ft]) above the top of active fuel (TAF), with a time delay of 30 minutes. In this mode, the GDCS equalizing lines allow coolant from the suppression pool into the RPV to provide long-term inventory control.

To assess the equilibrium between the reactor decay heat and the condensate flow rate from the PCCS, in RAI 6.3-33, the staff requested additional information regarding the normal and postaccident water level in the GCDS pool. RAI 6.3-33 was being tracked as a confirmatory item in the SER with open items. The applicant provided the information requested, and the staff concurs that the post-LOCA GDCS pool level depends on the type of pipe break and the break elevation. The staff confirmed that this change was included in DCD Tier 2, Revision 6. Therefore, RAI 6.3-33 is resolved.

A perforated steel plate covers the GDCS pool opening to the drywell airspace to prevent debris from entering the GDCS pool. The holes in the perforated steel plate will be smaller than the orifice holes in the fuel support orifice. In addition, an intake strainer is provided at the suction line from the suppression pool to prevent debris from entering the RPV when the GDCS draws suction from the suppression pool. Section 6.2.1.7.3 of this report provides the staff's evaluation of the strainers.

As noted earlier, the GDCS also provides cooling water to the drywell floor during a hypothetical severe accident. Section 19.2 of this report contains the staff's evaluation of the severe accident mitigation features.

All piping in the GDCS is stainless steel and rated for reactor pressure and temperature. The RPV injection line and the equalizing line nozzles all contain integral flow limiters with a venturi shape for pressure recovery. The minimum throat diameters of the nozzles are 7.62 cm (3 in.) and 5.08 cm (2 in.), respectively. GEH states that the nozzle throat is long enough to ensure that the homogeneous flow model can be used in the LOCA analyses. In RAI 6.3-13, the staff asked GEH to provide additional information on the choked flow model in its LOCA analyses and the nozzle throat lengths for which it is applicable. The staff requested this information to address the applicability of the TRACG04 flow-choking computer model to the ESBWR RPV injection line and equalizing line nozzles. RAI 6.3-13 was being tracked as an open item in the SER with open items.

The applicant submitted the following additional information in response to RAI 6.3-13 S01:

- TRACG has a subcooled choking model applicable to small length-to-diameter (L/D) throat conditions. The model prediction comparisons to data include choked flow for both smooth and abrupt area changes (i.e., orifices), thus validating the model for small L/D.

- TRACG is qualified over a range of 0.0–8.68 L/D through direct comparison to test data. GEH provided a table of tests that contains the L/D for the pressure suppression test facility (PSTF) critical flow tests, Marviken, and the Edwards Pipe Tests used to qualify the TRACG critical flow model.

- GEH provided a table of L/Ds for break lines. The values of L/Ds of ESBWR break lines are within the ranges of the TRACG qualification database.

Recognizing that it is not possible to have continuous L/D values in the range of test data, the ESBWR break throat L/D values are within the range of tests used to qualify the TRACG code choking model. Based on the RAI response, the staff concludes that the TRACG model covers L/Ds for the ESBWR break lines. Based on the applicant's response, RAI 6.3-13 S01 is resolved.

A squib valve is installed on each GDCS line. The valve is leakproof during normal operation. After opening, the squib valves will remain fully open. This type of squib-actuated valve is smaller than the squib-actuated DPV that has been tested at full size. Section 3.9.6.3.2.4 of this report contains the staff's evaluation of squib valve tests.

A check valve is installed on each of the GDCS injection lines to the RPV, upstream of the squib-actuated injection valves. The check valve prevents backflow from the RPV to the GDCS, thereby mitigating the consequences of spurious GDCS squib-actuated valve operations. The check valve is classified as Quality Group A, seismic Category I, and ASME Code, Section III, Class 1. The MCR has a remote check valve position indication. The staff noted that the applicant changed the description of the check valves in DCD Tier 2, Revision 3, Section 6.3.2.7.2. To evaluate the changes to the check valves, the staff made the following requests in RAI 6.3-78:

- Describe the design differences between the old and new designs.

- Add the typical check valve figure in the DCD, as before.

- Confirm that the check valves used for injection and equalization are of the same design.

- Provide additional information to demonstrate that the core remains covered, considering the failure of GDCS check valves as the single active failure for design-basis LOCAs. Provide this information for the cases where reactor vessel pressure is higher than that of the GDCS and the check valve fails to close.

RAI 6.3-78 was being tracked as an open item in the SER with open items. The applicant submitted its response to RAI 6.3-78 regarding the GDCS check valve design and confirmed that the GDCS injection line and equalization line check valves are the same. Section 6.3.2.3.3 of this report provides a detailed evaluation. Based on the staff evaluation of the applicant's description of the valve design in the response, RAI 6.3-78 is resolved.

Automatic Depressurization System

The ADS is part of the ECCS; it depressurizes the reactor so the low-pressure GDCS can supply makeup coolant to the reactor. Depressurization is achieved through the sequenced operation of 10 SRVs and 8 DPVs. Initially, five SRVs open upon an ECCS signal to start reducing RPV pressure, followed by five more SRVs after a time delay of 10 seconds. The sequence continues with groups of DPVs opening after successive time delays, as follows: Group I (three DPVs), 50 seconds; Group II (two DPVs), 100 seconds; Group III (two DPVs), 150 seconds; and Group IV (one DPV), 200 seconds.

Using a combination of SRVs and DPVs to accomplish the ADS function provides diversity in the design. The design of the DPVs reduces components and maintenance, compared to SRVs. The use of DPVs also reduces the number of SRVs and the need for SRV maintenance, periodic calibration, and testing. In addition, since DPVs discharge into the drywell atmosphere, their use reduces the number of SRV discharge lines and quenchers in the suppression pool. The SRVs and DPVs and associated controls and actuation circuits are located or protected so that the consequential effects of an accident cannot impair their function. The ADS is designed to withstand the effects of flooding, pipe whip, and jet impingement. ADS components are also qualified to withstand the harsh environment postulated for DBAs inside containment, including temperature, pressure, and radiation. Section 3.11 of this report provides further details regarding environmental qualifications. The SRVs and DPVs are designed with flange connections to allow easy removal for maintenance, testing, or rebuilding. In addition, they are designed so that routine maintenance and inspection can take place at their installed locations. The squib valve is classified as Quality Group A, seismic Category I, and ASME Code, Section III, Class 1.

GEH successfully conducted full-size tests of the DPV to demonstrate its operation. Section 3.9.6.3.2.4 of this report contains the staff's evaluation of the DPV tests.

Each of the 10 ADS SRVs is equipped with a seismically qualified pneumatic accumulator and check valve. Normally, a high-pressure nitrogen supply system provides nitrogen gas to the SRV accumulators. Section 9.3.8 of this report contains the staff's evaluation of the high-pressure nitrogen supply system. The accumulators ensure that the valves can be opened following the failure of the gas supplying the accumulators. The accumulator capacity is sufficient to actuate the valve once at drywell design pressure and at least twice under accident conditions. The containment design pressure is approximately 310.3 kPaG (45 psig). At the beginning of the accident, the containment pressure is much lower than design pressure, and hence the valve can function twice. The DPVs are squib-actuated and are not dependent on accumulators. Thus, the applicant has met TMI Action Plan Item II.K.3.28 in NUREG–0737, which is equivalent to 10 CFR 50.34(f)(1)(x), with respect to BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for nonsafety-related equipment or instrumentation and accounting for normal expected air (or nitrogen) leakage through valves. Section 20.4 of this report contains the staff evaluation of Item II.K.3.28.

The SRVs and DPVs are sized such that vessel depressurization and cooldown are slow enough to prevent the system from exceeding vessel integrity limits. GEH performed a thermal analysis that considered the effect of blowdown. Because of the ESBWR unique design, depressurization is expected to be slower than in the current BWR operating reactors. The RPV and the containment are designed to maintain structural integrity during an ADS event. Thus, the applicant has met TMI Action Plan Item II.K.3.45 in NUREG–0737, which is equivalent to 10 CFR 50.34(f)(1)(xi), with regard to providing an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during a rapid BWR cooldown.

Isolation Condenser System

The ICS has four passive high-pressure loops, each containing a heat exchanger that condenses steam on the tube side. The steamline connected to the vessel is normally open, and the condensate return line is normally closed. During a LOCA, the condensate return valves open to initiate the ICS operation. The water volume in the condensate return line is credited in the LOCA analysis. Section 5.4.6 of this report provides the staff's evaluation of the ICS.

Similar to hydrogen accumulation in PCCS as described in Section 6.2.2.3, there is a potential for hydrogen accumulation in the IC tubes during post-LOCA conditions. To address this issue for IC, the applicant proposed to isolate the IC soon after IC injection. In response to RAI 6.2-202 S01 the applicant provided a design change where, upon the opening of any two DPVs, the ICS isolation valves are automatically signaled to close. Resolution of this RAI is discussed in Section 6.2.2.3 of this report.

Standby Liquid Control System

The SLCS also supplies the reactor with additional liquid inventory during a LOCA. The SLCS accomplishes this function by firing squib valves to inject boron solution from the two accumulator tanks pressurized by nitrogen. Section 9.3.5 of this report contains the staff's evaluation of the SLCS.

Qualification of Emergency Core Cooling System

The ECCS is designed to meet seismic Category I requirements, in accordance with Revision 3 of RG 1.29. The ECCS will be housed in structures designed to withstand seismic events, tornadoes, floods, and other phenomena, in accordance with the requirements of GDC 2. The ECCS equipment design complies with the guidance in Revision 3 of RG 1.26, regarding the quality group classifications and standards for water-, steam-, and radioactive-waste-containing components. The ECCS is protected against pipe whip and discharging fluids, in compliance with the requirements of GDC 4. In addition, the ECCS equipment meets the environmental qualification requirements of GDC 4 regarding operation under normal and accident conditions. Chapter 3 of this report discusses these aspects of the ECCS design.

The ESBWR is proposed as a single unit design, and therefore, GDC 5, which concerns the sharing of SSCs among units, is not applicable to the ESBWR design.

The ESBWR core remains covered during all anticipated operational occurrences and accident conditions. Therefore, the ESBWR ECCS meets the requirements of GDC 17, as it relates to the design of the ECCS having sufficient capacity and capability to ensure core cooling.

Sections 4.2, 4.6, and 9.3.5 of this report discuss GDC 27, as it relates to the reactivity control systems having a combined ability, in conjunction with poison added by the ECCS, to reliably control reactivity changes under postulated accident conditions, with an appropriate margin for stuck rods.

The GDCS, ICS, and SLCS provide abundant emergency core cooling, thus satisfying the requirements of GDC 35. All the ECCSs are designed to permit appropriate periodic inspection of important components, such as the heat exchanger, valves, water injection nozzles, and piping, to ensure the integrity and capability of the systems; thus, GDC 36 is satisfied. The design of the systems within the ECCS permit appropriate periodic pressure and functional testing, thus satisfying GDC 37.

Section 7.3 of this report evaluates the ECCS instrumentation and controls. Section 6.2 of this report discusses the periodic testing and leak rate criteria for those valves that will isolate the reactor system from the ECCS. Section 5.2.5 of this report discusses the detection of leaks from those portions of the ECCS within the primary coolant pressure boundary.

In RAI 5.4-43, the staff requested a clarification about the applicability of TMI-2 action item II.K.3.15 to the ESBWR design. In its response to RAI 5.4-43, GEH stated that, even though the ICS uses differential pressure transmitters to detect a possible pipe break, the ICS does not use steam-driven pumps. Thus, TMI Action Plan Item II.K.3.15, "Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems," is not applicable to the ESBWR. The staff agrees with the applicant that this item is not applicable and hence the RAI 5.4-43 is resolved.

Manual activation of the ADS was originally required to provide adequate core cooling for transient and accident events that did not directly produce a high-drywell-pressure signal (e.g., stuck-open relief valve or steamline break outside containment), and that were further complicated by the loss of all high-pressure systems.

However, TMI Action Plan Item II.K.3.18 required all BWRs to modify their ADS actuation logic to eliminate the need for manual activation to ensure adequate core cooling. Instead, the ESBWR ADS equipment is activated automatically upon receipt of a signal of persistent low reactor water level with a delay of 10 seconds or sustained high drywell pressure with a delay of 60 minutes, without the need for operator action. Manual actuation is also possible. ADS complements manual actuation. RAI 6.3-10 requested additional information on the ADS control logic used to model the Level 1 setpoint in TRACG. RAI 6.3-10 was being tracked as an open item in the SER with open items. The applicant submitted its responses to RAIs 6.3-10 S01 and S02. The applicant provided detailed ADS actuation logic. The ADS will be initiated when the water level reaches Level 1 (11.5 m [37.7 ft] from RPV bottom). The safety margins for LOCAs, SBO, and loss of feedwater are well maintained with this setpoint, as demonstrated in RAI 6.3-10 S01, and DCD Tier 2, Sections 6.3, 15.2, and 15.5. In addition, the response to RAI 6.3-10 S02, clarified how the water level is calculated in the TRACG model. Section 6.3.2.3.5 of this report contains further evaluation. Based on the applicant's response, RAI 6.3-10 is resolved. Since the ADS logic includes a drywell high-pressure signal with a time delay, TMI Action Plan Item II.K.3.18 is satisfied.

Preoperational Tests

Preoperational tests will ensure the proper functioning of controls, instrumentation, pumps, piping, and valves. The applicant will measure pressure differentials and flow rates for later use in determining acceptable performance in periodic tests. Section 14.2 of this report notes the applicant's commitment to conformance to the guidelines in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 3, March 2007 for preoperational and initial startup testing of the ECCS.

Safe Shutdown

Establishing a safe-shutdown condition requires maintaining the reactor in a subcritical condition and adequate cooling to remove residual heat. One of the functional requirements for the ESBWR is that the plant can be brought to a stable condition using the safety-grade systems for all events. The Commission, in a staff requirements memorandum dated June 30, 1994, approved the position proposed in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Nonsafety Systems in Passive Plant Designs," dated March 28, 1994. This position accepts temperatures of 215.6 degrees C (420 degrees F) or below, rather than the cold shutdown (less than 93.3 degrees C [200 degrees F]) temperature specified in SRP 5.4.7, Branch Technical Position RSB 5-1, "Design Requirements of the RHR System," Draft Rev.4, April 1996, as the safe, stable condition that the passive decay heat removal system must be capable of achieving and maintaining following non-LOCA events. The SLCS establishes safe shutdown by providing the necessary reactivity control to maintain the core in a subcritical condition and by providing residual heat removal capability to maintain adequate core cooling. DCD Tier 2, Revision 9, Section 7.4, discusses the systems required for safe shutdown.

For all events, the ECCS will use the following systems to keep the reactor in a stable condition:

- ICS
- SLCS
- SRVs
- DPVs
- GDCS
- PCCS

The passive ICS automatically initiates upon closure of the MSIVs to remove decay heat following scram and isolation, and ICS condensate flow provides initial reactor coolant inventory makeup to the RPV. When the water reaches Level 1 in the reactor, the ADS, with operation of the SRVs and DPVs, initiates to depressurize the RPV.

Post-72-Hour Actions

The ESBWR passive decay heat removal systems are capable of achieving and maintaining safe, stable conditions for at least 72 hours without operator action following LOCAs. The IC and PCCS expansion pools have an installed capacity that provides at least 72 hours of reactor decay heat removal capability. Replenishing the IC and PCCS expansion pool inventory allows the heat rejection process to continue indefinitely. A safety-related independent FAPCS makeup line adds makeup water to the IC and PCCS expansion pools. A dedicated diesel-driven makeup pump system is connected to the FAPCS. This connection enables the site FPS to fill the upper IC and PCCS pools. This is acceptable because it complies with the guidelines in SECY-94-084.

Mechanical and Electrical Separation

The staff reviewed the ECCS design to confirm that the system's mechanical and electrical separation criteria are satisfied. Although a common tie exists between the ICS and DPVs on the stub line from the reactor vessel, there is no safety impact resulting from the cross-tie between the ICS and the DPVs. The GDCS Divisions B and C injection lines both connect to a common GDCS pool. This exception is acceptable, since there is sufficient redundancy in the GDCS. In response to RAI 6.3-12 S01, GEH provided a draft paragraph for inclusion in the DCD to clarify the mechanical separation provided in the design. Since there is adequate separation between the GDCS systems, the response is acceptable. RAI 6.3-12 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included this change in Revision 6 of the DCD. Based on the applicant's response, RAI 6.3-12 is resolved

System Reliability

The ESBWR ECCS is designed to satisfy a variety of requirements to ensure the availability and reliability of its safety functions, including redundancy (e.g., for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the design provides protection against single active and passive component failures; spurious failures; physical damage from fires, flooding, missiles, pipe whip, and accident loads; and environmental conditions, such as high-temperature and containment flood-up. The design reliability assurance program will include all risk-significant SSCs, as described in Section 17.4 of this report.

Generic Issues Related to the ECCS

Staff evaluation of TMI Action Plan Item III.D.1.1 is included in Section 20.4 of this report.

Since the ESBWR design does not include core spray or LPCI systems that can restart after a LOCA, TMI Action Plan Item II.K.21 is not applicable.

Inspections, Tests, Analyses, and Acceptance Criteria

DCD Tier 1 contains the ESBWR ITAAC, and Section 14.3 of this report includes the staff's evaluation of them.

In RAI 6.3-18, the staff asked GEH to provide the pool inventory in the ITAAC and a physical elevation inspection of the GDCS pool level. This RAI was being tracked as an open item in the SER with open items. In response in DCD Tier 2, GEH revised GDCS ITAAC Table 2.4.2.3 to include verifying the minimum drainable water volume and minimum water levels in GCDS pools. Based on the applicant's response, RAI 6.3-18 is resolved.

6.3.2 Emergency Core Cooling System Performance Analysis for Loss-of-Coolant Accident

6.3.2.1 *Regulatory Criteria*

DCD Tier 2, Section 6.3.3.7, presents the design bases for the ESBWR ECCS and the LOCA ECCS performance analysis. The staff based its review of the ECCS performance for the LOCA on information in DCD Tier 2, Revision 3; responses to RAIs; and topical reports referenced by the applicant. The staff conducted its evaluation in accordance with the requirements of 10 CFR 50.46 and the guidelines provided by SRP Section 6.3 and Section 15.6.5, Revision 3, issued 6/96. The approved LTR NEDC-33083P-A and its safety evaluation contain a detailed discussion of regulatory criteria.

6.3.2.2 *Summary of Technical Information*

6.3.2.2.1 Evaluation Model

GEH used the staff approved TRACG code, (See NEDC-33083P-A), to evaluate the ESBWR system response during a LOCA. Section 21.6 of this report summarizes the staff's evaluation of the TRACG code as applied to the ESBWR.

6.3.2.2.2 Uncertainty Analysis

On September 20, 2005, GEH provided a conference call summary with the NRC regarding the TRACG LOCA SER confirmatory items (See ADAMS Accession Number ML052910378). GEH stated that, since there is no core heatup, an uncertainty analysis on the PCT would not provide useful results. GEH stated that a bounding evaluation for the minimum water level in the chimney during a LOCA demonstrates that there is margin to core uncovery and heatup.

6.3.2.2.3 Failure Mode Analysis

As discussed in Section 6.3.1.2 of this report, in case of a LOCA, the GDCS, in conjunction with the ADS, the ICS, and the SLCS, provides the emergency core cooling. In DCD Tier 2, Revision 3, and in response to RAI 6.3-46, GEH analyzed eight LOCAs using the failure of one GDCS valve, one SRV, or one DPV. DCD Tier 2, Table 6.3-1, which is replicated below, identifies the most limiting combinations. Based on the applicant providing applicable information found in Table 6.3-1, RAI 6.3-46 is resolved.

Table 6.3-1. Single-Failure Evaluation.

Assumed Failure	Systems Remaining
1 DPV	10 SRVs, 7 DPVs, 3 ICSs, 2 SLCS accumulators, and 4 GDCSs with 8 injection lines
1 SRV	9 SRVs, 8 DPVs, 3 ICSs, 2 SLCS accumulators, and 4 GDCSs with 8 injection lines
1 GDCS injection valve	10 SRVs, 8 DPVs, 4 ICSs, 2 SLCS accumulators, and 4 GDCSs with 7 injection lines

6.3.2.2.4 Loss of Offsite Power

GEH analyzed the LOCAs with a loss of offsite power (LOOP) occurring at the same time as the initiation of the break.

6.3.2.2.5 Break Spectrum

Table 6.3-2 below shows all of the connections to the ESBWR RPV.

Table 6.3-2. ESBWR RPV Penetrations.

Piping Connection	Number of Lines	Elevation (relative to bottom of the vessel)	Break Area	Notes
Main Steamlines	4	22.84 m (74.93 ft)	0.09832 m^2 (1.058 ft^2)	Limited by venturi throat area
DPV/IC (DPV Stub Tube)	4	21.91 m (71.88 ft)	0.08320 m^2 (0.8956 ft^2)	Limited by venturi throat area (0.41 m [16-in.] Schedule 160 pipe)
Feedwater Nozzle	6	18.915 m (62.06 ft)	0.07420 m^2 (0.7986 ft^2)	Limited by feedwater nozzle area
RWCU/SDC Suction Line	2	17.215 m (56.48 ft)	0.06558 m^2 (0.7059 ft^2)	0.30 m(12-in.) Schedule 80 pipe
IC Drain Line	4	13.025 m (42.73 ft)	0.01824 m^2 (0.1963 ft^2)	Limited by venturi throat area (0.15 m [6-in.] diameter)

Piping Connection	Number of Lines	Elevation (relative to bottom of the vessel)	Break Area	Notes
GDCS Injection Lines	8	10.453 m (32.29 ft)	0.004561 m^2 (0.04910 ft^2)	Limited by venturi throat area (7.62 cm [3-in.] diameter)
SLCS Injection Line	2	9.709 m (31.85 ft)	0.000453 m^2 (0.0049 ft^2)	Limited by the nozzle area at the shroud penetration
GDCS Equalizing Line	4	8.453 m (27.73 ft)	0.002026 m^2 (0.02181 ft^2)	Limited by venturi throat area
RWCU/SDC Drain Line (bottom head drain line)	4	0.0 m (0.0 ft)	0.004052 m^2 (0.04361 ft^2)	Area of 2 nozzles (5.08 cm [2-in.] diameter)

The values in the above table are from Table 6.3-47-1 in the applicant's response to RAI 6.3-47. In RAI 6.3-47, the staff requested the applicant to include a table in the DCD to show the ECCS line break sizes and elevations. This RAI was being tracked as an open item in the SER with open items. GEH did so, and the staff confirmed that the table is listed in DCD Tier 2, Revision 6, Table 6.3-5a. Based on the applicant's response, RAI 6.3-47 is resolved.

Two bottom drain lines join a common header. Staff requested in RAI 6.3-58 that GEH provide the diameter of the lines at the vessel penetration and the diameter of the common header. In response to RAI 6.3-58, GEH stated that the penetration of the bottom drain line to the vessel is 50.8 mm (2 in.). Although the break area for the common header is larger than that of the two drain line nozzles, the flow is choked at the vessel penetrations, and GEH therefore assumes the area of the break to be the size of two of the nozzles. Based on the bottom drain line sizing information provided by the applicant, RAI 6.3-58 is resolved. GEH selected a representative set of cases to evaluate the spectrum of postulated break locations to demonstrate the ECCS performance. Specifically, GEH analyzed the following break locations, each with various single failures:

- MSL inside containment
- FWL
- GDCS injection line
- Bottom head drain line

The largest possible line breaks for the ESBWR are the DPV stub tube break, MSLB, FWLB, and RWCU/SDC suction line break. The DPV stub tube break will also include backflow through the IC return line; similarly, the total RWCU/SDC suction line break flow includes flow through the bottom head drain line. GEH analyzed the maximum inside steamline break and the maximum FWLB as representative cases for these four break locations. After an IC return line break, the ESBWR will rapidly depressurize through the ADS valves. Therefore, the results for this case are similar to those for the large steamline break case. For small line breaks, GEH analyzed the GDCS injection line break and the bottom head drain line breaks.

In RAI 6.3-46, the staff requested the applicant to provide the technical bases for the selection of the most limiting break size cases. In response, GEH submitted the minimum water level results for the following additional break locations:

- GDCS equalizing line
- DPV stub tube (DPV/IC steamline)
- RWCU/SDC return line
- IC drain line

GEH stated that the limiting cases are the GDCS injection line and IC drain line breaks. The applicant's results do not show heatup or core uncovery for any of the analyzed LOCAs. Since the acceptance criteria for 10 CFR 50.46 are not challenged for this event, GEH uses minimum static head in the chimney as a metric to determine the most limiting break. Staff provides Table 6.3-3 below that shows the various break scenarios analyzed by GEH and the minimum static head in the chimney during each event. Based on the information provided in the table, RAI 6.3-46 is resolved.

Table 6.3-3. Nominal ESBWR LOCA Calculations.

Break Location	Break Size m^2 (ft^2)	Single Failure	Minimum Chimney Static Head m (ft)
Steamline Inside Containment	0.09832 (1.058)	1 SRV	8.47 (27.8)
Steamline Inside Containment	0.09832 (1.058)	1 GDCS Valve	8.36 (27.43)
Steamline Inside Containment	0.09832 (1.058)	1 DPV	8.76 (28.74)
Feedwater Line	0.07420 (0.7986)	1 SRV	8.37 (27.47)
Feedwater Line	0.07420 (0.7986)	1 GDCS Valve	8.26 (27.09)
Feedwater Line	0.07420 (0.7986)	1 DPV	8.35 (27.3)
GDCS Injection Line	0.004561 (0.04910)	1 SRV	8.69 (28.52)
GDCS Injection Line	0.004561 (0.04910)	1 GDCS Valve	8.9 (29.19)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	8.73 (28.64)
Bottom Head Drain Line	0.004052 (0.04361)	1 SRV	8.35 (27.39)
Bottom Head Drain Line	0.004052 (0.04361)	1 GDCS Valve	8.62 (28.29)
Bottom Head Drain Line	0.004052 (0.04361)	1 DPV	8.42 (27.63)
ICS Drain Line	0.01824 (0.1963)	1 SRV	8.40 (27.55)
ICS Drain Line	0.01824 (0.1963)	1 GDCS Valve	8.55 (28.04)
ICS Drain Line	0.01824 (0.1963)	1 DPV	8.56 (28.08)

The values in the above table are from DCD Tier 2, Revision 9, Table 6.3-5.

GEH used nominal plant calculations to obtain the minimum chimney static head measurements set forth in Table 6.3-3 of this report. GEH did not perform an uncertainty analysis on the minimum chimney static head. Instead, it performed a bounding calculation on the two most limiting break locations—the ICS drain line break and the GDCS injection line break. The staff previously reviewed and approved the bounding assumptions, as documented in Section 2.7.2.1 of NEDC-33083P-A. Table 6.3-4 below presents the results of the applicant's calculations.

Table 6.3-4. Bounding ESBWR LOCA Calculations.

Break Location	Break Size m^2 (ft^2)	Single Failure	Minimum Chimney Static Head m (ft)
ICS Drain Line	0.01824 (0.1963)	1 SRV	8.33 (27.33)
ICS Drain Line	0.01824 (0.1963)	1 GDCS Valve	8.19 (26.87)
ICS Drain Line	0.01824 (0.1963)	1 DPV	8.31 (27.26)
GDCS Injection Line	0.004561 (0.04910)	1 SRV	8.82 (28.93)
GDCS Injection Line	0.004561 (0.04910)	1 GDCS Valve	8.34 (27.36)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	8.87 (29.09)

The values in the above table are from DCD Tier 2, Revision 7, Table 6.3-5.

6.3.2.2.6 Evaluation Model Parameters and Assumptions

GEH chose the evaluation model parameters and assumptions discussed below.

Initial Power Level DCD Tier 2, Table 6.3-11, states that GEH is using a core power of rated +2 percent for its bounding LOCA analysis.

Maximum Linear Heat Generation Rate DCD Tier 2, Table 6.3-11, states that GEH is using a peak linear heat generation rate of 44.8 kilowatts per meter (kW/m) (13.7 kW per foot [kW/ft]) for its bounding LOCA analysis.

Axial Power Shapes The applicant's TRACG model uses 35 axial nodes, with 32 representing the heated section of the channel. In NEDC-33083P-A, GEH stated that it is using a bottom peaked axial power shape. GEH does not perform the analysis with other power shapes.

Initial Stored Energy GEH assumes constant gap conductance throughout the LOCA. GEH uses these gap conductances as inputs into the TRACG code and calculates them through the GSTRM fuel mechanical code. Section 4.2 of this report discusses the applicability of the GSTRM code to the ESBWR. The applicant's fuel thermal conductivity model is based on that used in the PRIME03 code. RAI 6.3-54 and RAI 6.3-55 asked GEH to address the applicability of the PRIME03 code to the ESBWR. After receiving responses to RAI 6.3-54 and RAI 6.3-55, the staff issued RAI 6.3-54 S01, asking for experimental evidence of the data provided in response to RAI 6.3-54. Section 6.3.2.3.6 contains the staff's evaluation of the response.

Control Rod Insertion GEH uses a scram time delay with each LOCA case analyzed. DCD Tier 2, Table 6.3-1, states that the events are analyzed with a 2-second scram delay time. DCD Tier 2, Table 6.3-11, states that GEH is using the 1994 ANS decay heat standard.

Boric Acid Precipitation Boric acid will be injected into the RPV bypass during a LOCA as part of the SLCS initiation. GEH does not consider boric acid precipitation as part of short-term or long-term core cooling.

Containment Pressure Response Section 6.2 of this report discusses the containment pressure response.

ECCS Strainer Performance Evaluation Section 6.2.1.7.3 of this report discusses the ECCS strainer performance evaluation.

6.3.2.2.7 Reactor Protection and Emergency Core Cooling System Actions

The following sections give a narrative description of the sequence of events for the break locations presented in DCD Tier 2, Revision 9, and the ESBWR ECCS and RPS response.

Gravity-Driven Cooling System Line Break In DCD Tier 2 the GDCS line break with the failure of one injection valve is the most limiting. GEH showed the results of the TRACG analysis of this break in DCD Tier 2, Revision 9, Section 6.3.3.4. DCD Tier 2, Revision 9, Table 6.3-9, contains the operational sequence of the RPS and ECCS actions.

DCD Tier 2, Revision 9, Figures 6.3-32a and 6.3-32b, show the static head in the chimney and the two-phase level in the chimney. During the first 30 seconds after the break, because of flashing, the collapsed chimney level increased relative to the bottom of the chimney.

The system reaches the Level 2 setpoint approximately 15 seconds after the break. The MSIV will either close after the 30-second delay or on a low MSL pressure signal plus delay time. For the GDCS line break, the system first reaches the low MSL pressure setpoint at around 17 seconds, and the MSIVs close about 1 second later. DCD Tier 2, Revision 9, Figure 6.3-35b, shows that, at this point, the flow in the steamline goes to zero, and DCD Tier 2, Revision 9, Figure 6.3-34b, shows that the RPV pressure decrease slows.

DCD Tier 2, Figure 6.3-35b, shows that the break flow decreases at about 8 seconds. This is a result of the swell in the downcomer and of the break flow reaching the saturation temperature, when it begins voiding. In RAI 6.3-69, the staff requested the applicant to include figures of void fraction versus time for the break flow for the breaks presented in DCD Tier 2, Section 6.3. GEH showed a plot of the void fraction of the break flow in Figure 6.3-69-4a of the applicant's response to RAI 6.3-69. GEH stated, in response to RAI 6.3-69, that it would add these figures to the DCD. This plot shows that the void fraction increases until about 24 seconds, when the MSIVs close. At this time, the voids begin to collapse. The break flow void fraction is reduced to zero at about 30 seconds. At this time, the downcomer two-phase level begins to nearly equal that of the collapsed level in the downcomer. RAI 6.3-69 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included this change in Revision 6 of the DCD. Since TRACG adequately simulates voids fraction vs. time behavior as expected in the accident scenario, RAI 6.3-69 is therefore resolved.

The IC drain valves open on the LOOP. The drain valves open after a 15-second delay. DCD Tier 2, Figure 6.3-37b, shows the IC drain flow, which peaks early because of the additional water inventory in the IC drain tanks. There is a high flow rate until about 55 seconds, which is the time it takes for the IC drain tanks to empty. IC flow then decreases and begins to oscillate. This flow is from the IC steam condensation. In RAI 6.3-74, the staff requested the applicant to explain the cause of the increase in steam flow at about 200 seconds for all breaks. In response to RAI 6.3-74, GEH explained that, 180 seconds into the transient, there is a drop in the IC drain line water level, which causes an abrupt increase in the IC steam flow rate to fill the voided volume.

Since the applicant adequately explained the steam flow changes during the accident scenario, the response is acceptable and hence RAI 6.3-74 is resolved.

The system reaches the Level 1 setpoint approximately 220 seconds after the break. Level 1 must persist for 10 seconds to be confirmed. ADS initiation criteria are met at approximately 231 seconds after the break, and SRV actuation begins. DCD Tier 2, Figure 6.3-35a, shows the increase in steam flow at this time, resulting from the opening of the ADS valves. DCD Tier 2, Figure 6.3-36a, shows the steam flow contribution separately from the SRVs and DPVs and also from the IC. The downcomer flashes when ADS actuation begins. DCD Tier 2, Figure 6.3-33a, shows this as an increase in the two-phase level. DCD Tier 2, Figure 6.3-32a, shows that the collapsed chimney level decreases at this time. The collapsed level oscillates with each SRV and DPV actuation, then, steadily decreases to its minimum at around 500 seconds into the transient.

The SLCS timer begins when the system reaches the Level 1 setpoint. The SLCS timer times out in 50 seconds, at the same time as DPV actuation, and the SLCS actuates at about 281 seconds, as shown in DCD Tier 2, Figure 6.3-37a.

The GDCS timer is also initiated with the Level 1 setpoint. The GDCS timer times out in 150 seconds, and the GDCS injection valves open at 380 seconds after the initiation of the break. DCD Tier 2, Figure 6.3-35a, shows the GDCS pool in the broken line starting to empty into the drywell at this time. However, the RPV pressure is still too high for the other GDCS trains to inject into the RPV.

DCD Tier 2, Figure 6.3-37a, shows the GDCS beginning to inject at about 488 seconds into the event, when the RPV pressure decreases to that of the GDCS. There is a spike in flow at the onset of GDCS initiation. This shows the steam from the RPV colliding with the subcooled GDCS flow. The collapsed level in the chimney begins to rise from its minimum value after the GDCS injection begins.

The collapsed chimney level continues to recover as a result of the GDCS injection. At about 1450 seconds into the transient, the level starts to experience large oscillations. GEH states, in NEDC-33083P-A, that these are manometric oscillations. The collapsed and two-phase chimney levels continue to oscillate. However, on average, the level continues to recover until it reaches the top of the chimney partitions.

The staff also requested in RAI 6.3-68 that GEH provide reactor power as a function of time and in RAI 6.3-70, requested the applicant to state if the cases presented in the DCD are run with nominal or bounding conditions. RAI 6.3-70 was being tracked as an open item. DCD Tier 2, Figure 6.3-39, includes the reactor power as a function of time plots, and it clearly labeled each plot with the nominal or bounding conditions. Based on the applicant's response, both RAI 6.3-70 and RAI 6.3-68 is resolved.

Main Steamline Break DCD Tier 2, Revision 9, Figures 6.3-15a through 6.3-22b, describe the system response of an MSLB inside containment with one GDCS valve failure. DCD Tier 2, Revision 9, Table 6.3-8, sets forth the operational sequence of the RPS and ECCS actions. The RPS and ECCS response is similar to the GDCS line break discussed above. One of the main differences between the responses of the MSLB and the GDCS line break is that the MSLB depressurizes much faster than the GDCS line break because of the larger break size. In addition, the Level 1 setpoint actuates later because of the level swell.

Feedwater Line Break DCD Tier 2, Revision 9, Figures 6.3-7a through 6.3-14b, describe the system's response to an FWLB with one GDCS valve failure. DCD Tier 2, Revision 9, Table 6.3-7, shows the operational sequence of the RPS and ECCS actions. The RPS and ECCS response is similar to the GDCS line break discussed above. One of the main differences between the FWLB and the GDCS line break is that the FWLB depressurizes faster than the GDCS line break because of the larger break area. Also, the higher elevation causes the FWLB to respond more like the steamline break, once the two-phase level drops below the elevation of the feedwater sparger. Similar to the MSLB, because of the level swell, the Level 1 setpoint actuates later than the GDCS line break but sooner than the MSLB.

Bottom Drain Line Break DCD Tier 2, Revision 9, Figures 6.3-23a through 6.3-30b, describe the system response of a bottom drain line break with one GDCS valve failure. DCD Tier 2, Revision 9, Table 6.3-10, shows the operational sequence of the RPS and ECCS actions. The RPS and ECCS response is similar to the GDCS line break discussed above. The bottom drain line break has a lower elevation and a smaller break area than the GDCS line break. Hence, the vessel depressurizes more slowly, and the Level 1 setpoint actuates later than in the GDCS line break.

6.3.2.2.8 Long-Term Core Cooling

In a letter dated October 6, 2005 (See ADAMS Accession Number ML053140221), GEH submitted details on long-term core cooling. This letter included a discussion of long-term inventory distribution for four break locations—(1) MSLB, (2) FWLB, (3) bottom drain line break, and (4) GDCS line break. GEH based these analyses on DCD Revision 0 and updated them when it submitted the responses to RAI 6.3-64 and RAI 21.6-98. In RAI 6.3-64, the staff requested the applicant to submit the plots of the core level demonstrating that the core will remain covered for 72 hours for the limiting break. In response to RAI 6.3-64, GEH submitted a long-term core cooling analysis for the GDCS line break with one DPV failure. The staff asked GEH for additional information on this analysis. In RAI 6.3-79, GEH pointed out that it discusses long-term cooling. The response discussed TRACG calculation results for the short-term (0-2,000 s) and long-term (0-72 hours) core cooling. The discussion showed that, for all break locations, the water levels are above the reactor core and above the GDCS equalization line water injection setpoint Level 0.5 for 30 days. Section 6.3.2.3.1 of this report contains the evaluation of the response to RAI 6.3-79. Section 21.6 of this report discusses the closure of RAI 21.6-98. The staff received a response to RAI 6.3-64 S01, in April 2008, and Section 6.3.2.3.8 of this report discusses its evaluation and subsequent resolution.

Long-Term Core Cooling for Main Steamline Break In the long-term MSLB, the GDCS will drain to the level of the break (i.e., the DPVs), which will leave about two-thirds of the GDCS inventory in the pools. The PCCS will condense the steam generated by decay heat and return it to the vessel through the GDCS. Some steam will condense on the drywell surfaces and not return to the RPV, leaving a small amount in the lower drywell.

Long-Term Core Cooling for Feedwater Line Break The long-term core cooling for the FWLB is similar to the MSLB, because it is a higher-elevation break. The GDCS pools will drain down to the level of the FWL sparger, which is close to the bottom of the elevation of the GDCS pools. There is a period of time when the GDCS pools are drained, and the PCCS does not condense the steam at the same rate as the decay heat power generated by the core, and so the level in the downcomer decreases at a faster rate. RAI 6.3-64 and RAI 21.6-98 addressed concerns related to long-term core cooling. RAI 21.6-98 is discussed in Section 21.6 of this report. Some steam will condense on the drywell surfaces and not return to the RPV, leaving a small amount in the lower drywell. The level in the drywell gets high enough to return to the suppression pool through the spillover holes in the vertical vent pipes. However, the drywell level remains well below the RPV break location in the FWL sparger.

Long-Term Core Cooling for Bottom Drain Line Break Since this break is on the bottom of the vessel, the inventory in the lower drywell becomes important. The GDCS pool empties in a few hours into the event, and the level in the downcomer begins to decrease at a faster rate. The drywell fills up to the elevation of the spillover hole (between the suppression pool and the drywell) at about 5 hours. The level in the downcomer and the RPV goes below the top of the chimney partitions about 6.5 hours into the event and continues to drop until the level reaches that of the spillover hole. The elevation of the spillover hole is several feet above the bottom of the reactor vessel, which is approximately 10 m (32.8 ft) above the TAF. At about 8 hours into the event, the levels in the drywell, RPV, and downcomer remain nearly constant at the spillover hole level. The PCCS maintains the levels by condensing the steam from decay heat and returning it to the vessel through the GDCS.

Long-Term Core Cooling for Gravity-Driven Line Break The long-term behavior of this break is similar to that of the bottom drain line break described above, in that, once the GDCS pool drains and the levels in the downcomer and the RPV start to fall, they will level out at the spillover hole elevation because the GDCS injection line is below that of the spillover hole. Since the GDCS line is broken, more inventory enters the drywell earlier in the event, and the level in the drywell reaches that of the spillover holes at about 3 hours into the event. Also, since the GDCS pools lose inventory faster because of the broken line, the GDCS pools empty at about 4 hours into the event.

6.3.2.3 *Staff Evaluation*

6.3.2.3.1 Evaluation Model

The staff reviewed and approved the GEH evaluation model (TRACG) for the 4000 MWt ESBWR design, described in NEDC-33083-A. Section 21.6 of this report provides an evaluation of its applicability to the current ESBWR 4500 MWt design. The ESBWR LOCA analyses show that the core does not heat up or uncover. Therefore, the staff did not review or approve the use of TRACG for core heatup or uncovery. The staff's acceptance of the ECCS performance for the ESBWR is based on maintaining a static head of water above the TAF.

6.3.2.3.2 Uncertainty Analysis

Regulations in 10 CFR 50.46(a)(1)(i) state, in part, that "comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for." Furthermore, 10 CFR 50.46(a)(1)(ii) states, "Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models." The staff issued RAI 6.3-81 to request that GEH demonstrate how the LOCA analyses comply with these requirements. RAI 6.3-81 was being tracked as an open item in the SER with open items. In response to RAI 6.3-81, GEH stated for ESBWR LOCAs, that because there is no core uncovery and no core heatup, a statistical analysis of the PCT does not serve any useful purpose. The best-estimate PCT and the 95/95 PCT would both be close to the saturation temperature corresponding to the peak steam dome pressure reached in the accidents. In the case of ESBWR LOCAs, there is a margin of over 871 degrees C (1,600 degrees F) to the limit of 1,204 degrees C (2,200 degrees F, the acceptance criterion set forth in 10 CFR 50.46(b)). GEH further stated that the static head inside the chimney (in meters of water) is selected as the figure of merit for comparison and to evaluate the impact of uncertainties in model parameters and plant parameters. This collapsed level is defined as the equivalent height of water corresponding to the static head of the two-phase mixture above the top of the core. The TRACG model parameter uncertainties and plant parameter uncertainties have been identified (NEDC-33083P-A, Sections 2.4 and 2.5.3). GEH performed sensitivity studies by varying each of these parameters from the lower bound to the upper bound value.

The impact on the chimney static head is between -0.3 m (-0.98 ft) to +0.2 m (+0.66 ft) (NEDC-33083P-A, Section 2.4.4.2), which is less than the minimum static head in the chimney from the parametric studies. Therefore, GEH proposed that a simple calculation be made setting the most significant parameters at the 2-sigma values to obtain a bounding estimate of the minimum level. The staff finds this approach acceptable and concurs that the ESBWR LOCA results demonstrate that there is a high probability that there is no core uncovery or heatup and that the PCT would be close to the saturation temperature corresponding to the peak steam dome pressure reached in the accident. The staff concludes that the GEH LOCA results comply with the requirements in 10 CFR 50.46. Based on the applicant's response, RAI 6.3-81 is resolved.

6.3.2.3.3 Failure Mode Analysis

GDC 35 requires that the ECCS be able to accomplish its function in the event of a single failure. In DCD Tier 2, Revision 9, Section 6.3.3, GEH provided an analysis to demonstrate the most limiting break size, break location, and single failure for the ESBWR. The staff reviewed the system description, process diagram, and ECCS performance analysis to ensure that the applicant considered all credible single active failures. The following sections describe the staff's evaluation of the single failures assumed in the analyses for each of the credited ECCSs.

SRP Section 6.3 states that the long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. In RAI 6.3-79, the staff requested the applicant to clarify whether the ESBWR design takes credit for any passive component during long term post LOCA (i.e. beyond 72 hours) cooling. RAI 6.3-79 was being tracked as an open item in the SER with open items. In response, GEH stated that, for the ESBWR design, conformance to the requirement of adequate long-term cooling is assured and demonstrated for any LOCA where the water level can be restored and maintained at a level above the top of the reactor core. The response discussed TRACG calculation results for a short-term (0–2,000 s) and long-term (0–72 h) calculation. These calculations used assumptions with possible single failures of ECCS components. GEH then qualitatively determined, from the TRACG long-term (0–72 h) calculation, that the water level will remain near an equilibrium level above the core. For break locations lower than the spillover hole, the water level will remain at the final equilibrium level at the spillover hole level. For break locations higher than the spillover hole, GEH estimated, from the TRACG long-term (0–72 h) calculation, the inventory loss to the lower drywell caused by drywell wall condensation. The estimation showed that for all break locations the water levels are above the reactor core and above the GDCS equalization line water injection setpoint, Level 0.5, for 30 days. Furthermore, by design, if the RPV water level drops below Level 0.5, these equalization lines would be actuated. After actuation, these equalization lines provide the long-term post-LOCA makeup water to the RPV from the suppression pool. The suppression pool's normal water level is about 10 m (32.81 ft) from the RPV bottom, or 2.5 m (8.20 ft) above the TAF. The addition of the suppression pool water will ensure that the reactor core is covered at a level above the TAF for an indefinite long-term period. For these reasons, the staff concurs that the design provided adequate long-term cooling. RAI 6.3-79 is therefore resolved.

GDCS Single Failure In RAI 6.3-43, staff requested the applicant provide additional information on the single failure analyses for the GDCS. The GDCS consists of three pools and eight injection lines. In response, GEH provided additional information on the modeling of the GDCS. Staff followed up with RAI 6.3-43 S01 and requested the applicant to document the December 2006, audit discussions related to the comparison of different break/valve failure combinations and explanations that the applicant modeled the worst single failure. Since there are multiple GDCS lines and multiple GDCS pools, there are multiple combinations of failed valve and broken injection line combinations that are possible. GEH responded by updating DCD Tier 2, Chapter 6 with tables showing the modeling of the most limiting combination of break locations and valve failures. The staff confirmed that GEH chose the most conservative combination of valve failure and line break by reviewing the evaluation of initial injection flow rate and total long-term GDCS water volume. Therefore, RAI 6.3-43 is resolved.

The GDCS check valve must be closed upon initiation of the squib valves, since the RPV pressure is higher than that of the GDCS. In RAI 6.3-78, the staff requested that GEH evaluate the possibility of this failure, because it could result in additional coolant loss. In response, GEH stated that the old design was a biased-open, tilting disk check valve installed in a horizontal piping run. The new design is a normally open, piston check valve, installed in a horizontal or vertical piping run. DCD Tier 2, Revision 5, updated the GDCS check valve description. GEH confirmed that the GDCS injection and equalization line check valves are of the same design. GEH further stated that it added an ITAAC item to DCD Tier 1, Section 2.4.2, and Table 2.4.2-3. The ITAAC is to use GDCS check valve testing to measure the fully open flow coefficient in the reverse flow direction, and to verify that the measured value is less than the value assumed in the LOCA analyses. This verification will confirm that the check valve will function as designed and the core would remain covered in the event of a GDCS check valve failure following a LOCA, despite back flow through the GDCS injection line. Since the valves will be functionally tested during ITAAC, the staff accepted the GEH response and, therefore, RAI 6.3-78 is resolved. However, the staff did not see an analysis with a back flow in the GDCS drain line. The staff issued a new RAI 6.3-84, asking GEH to analyze cases in the event the GDCS check valve failed to close; this was an open item. In response, GEH provided calculation results for the limiting IC drain line break, where one of the GDCS check valves failed to close. The calculation showed that the reactor minimum level in the internal chimney during the LOCA would be 85.7 cm (33.74 in.) above the TAF. The calculation showed that the core would be covered with water, and the staff finds that the design would provide adequate cooling during this event. Therefore, RAI 6.3-84 is resolved.

GEH performed analyses of all design-basis LOCAs, assuming one GDCS squib valve fails to open. DCD Tier 2, Revision 3, Table 6.3-5, provides the results of the analyses, which show that the core remains covered for the GDCS line break, the MSLB, the bottom drain line break, and the FWLB with one GDCS injection line failure. Table 6.3-46-1, in the applicant's response to RAI 6.3-46, shows that the core remains covered for the GDCS equalizing line, the DPV stub tube (DPV/IC steamline), the RWCU/SDC return line, and the IC return line breaks with one GDCS injection line failure.

ADS Single Failure The ADS consists of DPVs and SRVs, as described in Section 6.3.2 of this report. GEH performed analyses of all design-basis LOCAs, assuming failure of either a DPV or an SRV. DCD Tier 2, Revision 3, Table 6.3-5, presents the results of the analyses, which show that the core remains covered for the GDCS line break, the MSLB, the bottom drain line break, and the FWLB, with one SRV or one DPV failure. Table 6.3-46-1, in the applicant's response to RAI 6.3-46, shows that the core remains covered for the GDCS equalizing line, the DPV stub tube (DPV/IC steamline), the RWCU/SDC return line, and the IC drain line breaks, with a failure of either a DPV or an SRV.

SLCS Single Failure Section 9.3.5 of this report provides the SLCS evaluation. This section shows that no single active failure of the SLCS can prevent either of the SLCS trains from injecting. Therefore, the staff finds that the applicant's assumption that the SLCS does not fail during any LOCA is acceptable and that the design of the SLCS complies with GDC 35, as it relates to ECCS performance.

One train of the SLCS will fail if there is a break in an SLCS line because inventory will be lost through the break. In RAI 6.3-65, the staff requested the applicant to evaluate the consequences of a break in the SLCS injection line with the worst single failure. In response, GEH showed that the collapsed liquid level in the downcomer does not drop to the Level 1 elevation and, therefore, does not initiate any ECCS during the first 2,000 seconds of the event and does not require SLCS injection. In RAI 6.3-65 S01, the staff requested that GEH discuss the long-term results of the SLCS line break. In response and providing an update in Revision 1, GEH provided a full analysis, using TRACG with an SLCS line break. The analysis showed that a late ADS open actuation caused by the smaller break size, compared to the other break scenarios, and the minimum water level is above the top of the active core. The applicant provided a sensitivity analysis, with and without ICS heat transfer modeling. With the ICS heat transfer modeling, the RPV pressure decreased more slowly, and this caused slower inventory loss. The calculation showed that the ADS initiated around 6,674 seconds. For the case without the ICS heat transfer modeling, after MSIV closure, the RPV pressure rose and reached the SRV setpoints. The SRV discharged RPV steam into the suppression pool, which resulted in more RPV inventory loss. The L1 ADS initiation setpoint is reached around 1,731 seconds. In both sensitivity cases, after ADS initiation, the GDCS recovered the water level. The staff finds that the calculation plots showed the water level is above the TAF, which is an indication that the ECCS has provided adequate cooling. Based on the applicant's response, RAI 6.3-65 is resolved.

ICS Single Failure GEH did not take credit for the heat removal capability of the ICS in DCD Tier 2, Revision 9, but modeled the inventory in the ICS drain tanks during a LOCA. The condensate drain valve for the ICS is single-failure-proof. There is a bypass valve that may be actuated in the event the condensate drain valve fails to open. Section 5.4.6 of this report describes this, and DCD Tier 2, Revision 3, Figure 5.1-3, depicts it. For all design-basis LOCA analyses, GEH always assumed that only three out of the four ICs are available during a LOCA, because one may be out of service. DCD Tier 2, Revision 9, Table 6.3-5, presents the results of the analyses, which show that the core remains covered for the GDCS line break, the MSLB, the bottom drain line break, and the FWLB with one inoperable IC.

In RAI 6.3-46 S01, the staff requested the applicant to submit the technical bases for the limiting break in the break spectrum analyses. The applicant's response to RAI 6.3-46 S01 shows that the core remains covered for the GDCS equalizing line, the DPV stub tube (DPV/IC steamline), and the RWCU/SDC return line breaks with one inoperable IC. For the IC drain line break, the IC that is out of service may be a different IC than the one attached to the broken line, so GEH assumed only two ICs were available for this event. The results in Table 6.3-46-1 show that the core remains covered for this event. The historical account and resolution of RAI 6.3-46 is described later in section 6.3.2.3.5.

In RAI 6.3-65 S01 (which is discussed further in section 6.3.2.3.5 of this report), the staff asked GEH to verify how many ICs were operating during the SLCS break evaluation. RAI 6.3-65 was being tracked as an open item in the SER with open items. GEH responded that there are four ICs associated with an ESBWR; however, the analysis of this event takes credit for only three of them. This resolved the availability of ICs. Based on the applicant's response, RAI 6.3-65 is resolved with regard to the availability of ICs. Complete RAI 6.3-65 resolution is described in previous paragraphs of section 6.3.2.3.3.

In RAI 6.3-66, the staff requested that GEH include a statement that the LOCA RPV level analyses take credit for the IC heat removal capacity and the CRD hydraulic control unit injection. In response, GEH stated that it will revise Table 6.3-1B.3 to include the drain line water inventory. The staff finds this approach acceptable. RAI 6.3-66 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that, in DCD Tier 2, Table 6.3-1, the analysis included the ICs and CRD water inventory and that GEH took no credit for IC heat removal in Table 6A-1. The applicant made appropriate DCD changes requested by the staff and therefore, RAI 6.3-66 is resolved.

<u>Vacuum Breaker Failure</u> There is a vacuum breaker between the drywell and the suppression pool that opens if the wetwell pressure exceeds that of the drywell. Failure of this valve to close after opening would cause steam to leak from the drywell to the wetwell bypassing the suppression pool at a rate higher than the design steam leakage value. Steam that enters the wetwell bypassing the suppression pool does not get condensed by the suppression pool and raises the wetwell pressure and eventually the drywell pressure. In DCD Tier 2, Revision 3, Section 6.2.1.1.2, GEH added an alternate means to close this opening by adding a vacuum breaker isolation valve (VBIV) that would allow the vacuum breaker system to remain operable with a single active failure of one vacuum breaker. The staff requested additional information about the block valve operation, control and its impact on containment and RPV analysis in RAI 6.3-63. RAI 6.3-63 was being tracked as an open item in the ESBWR SER with Open Items.

Staff reviewed the GEH response to RAI 6.3-63 and finds that this design approach is acceptable. See evaluation for DCD Tier 2, Section 6.2.1.1.2 under RAI 6.2-148 for further detail. RAI 6.3-63 is therefore resolved.

<u>Bottom Drain Line Isolation</u> The bottom drain line is open during normal operations for RWCU. In the event of a LOCA, it is possible that, if this line fails to isolate, additional loss of inventory may occur. In RAI 6.3-59, the staff requested the applicant to explain the signals which will isolate the bottom drain valves and the consequences if these valves were to fail to isolate during a LOCA. In response, GEH confirmed that there are two isolation valves in series; therefore, the failure of one valve to close would not result in a failure of the system to isolate. In addition, GEH provided the signals that would isolate the bottom drain line in the event of a LOCA. These signals include the following:

- Reactor vessel low water Level 2

- Reactor vessel low water Level 1
- MSL tunnel high ambient temperature
- High flow in the RWCU/SDC loop
- SLCS in operation

Based on this information, the staff finds that this system will be isolated during a LOCA and that there will be no additional inventory lost. Therefore, RAI 6.3-59 is resolved.

CRD Hydraulic Control Unit (HCU) In its analyses, GEH assumes that HCU inventory is injected into the vessel during a scram. GEH does not fail this injection source as part of its LOCA analyses. The volume of water injected into the vessel for one HCU is negligible, compared to the other ECCS sources, and its failure will not provide limiting results. In RAI 6.3-66, the staff requested the applicant to revise the DCD to include a statement that they take credit for the IC heat removal capacity and the water addition from the Hydraulic Control Unit (HCU). In response, GEH stated that it would include the HCU injection as DCD Tier 2, Table 6.3-1 B.6. RAI 6.3-66 was being tracked as a confirmatory item in the SER with open items. The staff verified the DCD was revised to include the credit for HCU. The response regarding the HCU credit in RAI 6.3-66 is acceptable. However, since HCUs are classified as Safety Class-2 in DCD Tier 2, Table 3.2-1, they are not considered to be safety grade. In RAI 6.3-87, the staff requested a justification for the use of HCU scram water in the LOCA analysis. GEH submitted a response to RAI 6.3-87. Also, the staff raised the CRD system classification issue in RAI 3.2-21. The applicant provided sufficient information in these responses justifying the classification and the qualification of the system. Therefore, RAIs 6.3-87 and 3.2-21 are resolved. Section 3.2.2.3.7 of this report discusses the staff's resolution of RAI 3.2-21 with regard to this issue.

Conclusion of Single-Failure Evaluation The staff examined failure possibilities and their significance. The GEH design selected single failures of one GDCS injection valve, one DPV, and one SRV for the LOCA analysis. The staff concurs that the failure of a DPV or SRV results in the greatest reduction in the depressurization rate from ADS actuation and results in a delay in GDCS injection. The failure of one GDCS injection valve results in the greatest reduction in the GDCS reflooding rate. The staff agrees with the discussion in the DCD and finds the single failure selection to be reasonable and acceptable.

6.3.2.3.4 Loss of Offsite Power

GDC 35 requires that the ECCS be able to accomplish its function in the event of a LOOP. To demonstrate that the ECCS performance meets the design requirements, GEH assumed a LOOP occurs coincident with the break for each of the design-basis LOCAs analyzed. This causes the reactor to scram and the ICS to initiate upon the loss-of-power signal. If there were no LOOP at the initiation of the break, there would be a delay in the actuation of these systems, as they would actuate on their own trip setpoints. GEH states that there is a loss of feedwater from a LOOP and assumes a loss of feedwater is more conservative than incorporating the delays. In DCD Tier 2, Revision 3, GEH changed the feedwater isolation to be safety-grade, and it is isolated upon a sensed differential pressure between the FWLs, coincident with high drywell pressure. The staff agrees with GEH that a LOOP, coincident with the break, is a conservative assumption because of the feedwater isolation. For the FWLB, the high-drywell-pressure signal occurs before 1 second (as shown in DCD Tier 2, Table 6.3-7) into the transient, meaning that the assumption of the loss of power at the break gives virtually the same scram and ICS response. GEH did not evaluate the effects of allowing the reactor and ICS to initiate on their own trip setpoints for a small-break LOCA. However, the staff agrees with GEH that the loss of feedwater during this time is a more reasonable assumption. Therefore, the staff finds the applicant's LOOP assumption to be appropriate for ESBWR LOCA analyses.

6.3.2.3.5 Break Spectrum

GEH showed the results of a LOCA at eight different break locations with three single failures. In each of these 24 cases, the core remains covered throughout the entire blowdown phase and through the reflood phase (until 2,000 seconds after the break). GEH uses minimum static head in the chimney as a metric to determine the most limiting break. The staff finds the labeling of DCD Tier 2, Revision 3, Table 6.3-5, misleading because GEH labels these values as "minimum chimney static head level above vessel zero," and calculates these values by collapsing the level in the chimney, not considering the void fraction in the core. In RAI 6.3-77, the staff requested that GEH change this label in the next revision of the DCD or else justify that it is the same (i.e., show that, when considering the void fraction in the core, the collapsed level remains the same). In response, GEH stated that it would update the language in the next revision of the DCD. GEH explained that the chimney static head level with reference to vessel zero is calculated by adding the equivalent height of water corresponding to the static head of the two-phase mixture inside the chimney to the elevation (7.896 m [25.91 ft]) of the bottom of the chimney. RAI 6.3-77 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included the above change in Revision 6 of the DCD. Therefore, RAI 6.3-77 is resolved.

GEH performed each of the 24 calculations using nominal conditions. GEH stated that the ICS drain line and the gravity injection line breaks are the most limiting cases and performed calculations for these two break locations, using bounding conditions. For these two cases, the core still remains covered. GEH was asked to clarify the justification for the limiting cases in Items A and C of RAI 6.3-46. In DCD Tier 2, Revision 5, GEH stated that the calculation results showed that the ICS drain line and GDCS return line break are the most limiting cases. Therefore, the staff had no further questions regarding RAI 6.3-46, Items A and C. Resolution of this RAI is discussed below.

RAI 6.3-86 asked GEH to show the sensitivity calculation results to demonstrate that the ICS drain line is the limiting case. GEH provided the sensitivity calculation results in its response. The nominal sensitivity calculation showed that the ICS drain line is the most limiting in terms of the chimney level. GEH further committed to documenting the sensitivity results in the new DCD revisions. The staff confirmed that the applicant included this change in Revision 6 of the DCD. Based on the applicant's response, RAI 6.3-86 is resolved.

However, there are still inconsistencies about limiting breaks in Revision 5 of the DCD, and RAI 6.3-85 requested a clarification from GEH. GEH responded with corrected DCD markups. . The response is satisfactory, and RAI 6.3-85 is resolved. RAI 6.3-83 asked GEH to provide consistent tables for the LOCA break sizes analyzed and the LOCA analysis results in Tables 6.3-5a and 6.3-5 of the DCD. GEH responded that it provided the non-limiting LOCA results in its response to RAI 6.3-46 and that the DCD documents contained the most significant LOCA results. Based on the applicant's response, RAI 6.3-83 is resolved.

In RAI 6.3-46, the staff requested the applicant to submit the technical bases for the limiting break in the break spectrum analyses. RAI 6.3-46 was being tracked as an open item in the SER with open items. In response, GEH performed a sensitivity study of the GDCS line break size. The break sizes for this study ranged from the full double-ended break to 80, 60, 40, and 20 percent of this size. GEH ran these cases using nominal conditions and the failure of one

GDCS injection valve. The 80-percent case gave the most limiting results. GEH then ran the cases for the 100-percent and 80-percent break sizes, using bounding assumptions. For these two cases, the 100-percent size is still the most limiting break for this location. In RAI 6.3-46 S01, the staff asked GEH to explain why this is so. In addition, the staff asked GEH why it did not evaluate the 60-, 40-, or 20-percent break sizes using bounding assumptions. In response, GEH stated it would provide a qualitative argument as to why very small breaks (i.e., smaller than 20 percent) are not limiting. In response GEH stated that the minimum water level difference between break sizes of 100 percent and 80 percent for bounding assumptions is about 0.01 m [0.4 in.], which is negligible. Therefore, there is no need to judge why a 100-percent break case is more limiting than an 80-percent break case. The staff accepted this argument, noticing the similarity of the system response for 80-percent and 100-percent breaks. GEH also stated, in its response to RAI 6.3-46, Item B, that, since the chimney static head for the 60-percent, 40-percent, and 20-percent nominal cases is higher than for the 100-percent and 80-percent cases, it is not necessary to analyze those cases with the bounding conditions. Therefore, the selection of bounding cases is acceptable. Based on the applicant’s response, RAI 6.3-46 is resolved.

GEH did not analyze a break in the SLCS injection line. The staff was concerned about this break since it would also cause the loss of an SLCS injection train. One train of the SLCS will fail if there is a break in an SLCS line because inventory will be lost through the break. In RAI 6.3-65, the staff requested the applicant to address the consequences of the SLCS line break. In response, GEH showed that the collapsed liquid level in the downcomer does not drop to the Level 1 elevation and therefore does not initiate any ECCS during the first 2,000 seconds of the event. In a RAI 6.3-65 S01, the staff requested additional information from GEH on the event after 2,000 seconds. Section 6.3.2.3.3 of this report contains the resolution of RAI 6.3-65.

Section 6.3.1.3 of this report describes staff’s request in RAI 6.3-10 about the ADS control logic used to model Level 1 setpoint in TRACG. RAI 6.3-10 was being tracked as an open item. In RAI 6.3-10 S01, the staff requested the applicant to explain in detail why the RPV Level 1.5 plus drywell high pressure and the Level 1.5 plus delay timer were removed from the ECCS initiation logic. In a relevant RAI, RAI 6.3-16, staff requested the applicant to clarify the DCD on GDCS initiation. The GEH response directed the staff to the DCD Tier 2, Table 6.3-1 where initiating signals and levels are listed. In a followup RAI 6.3-16 S01, the staff requested the applicant to provide the technical basis for the settings of the timer delays associated with the ECCS initiation logic. In response to both request for RAI 6.3-10 S01 and RAI 6.3-16 S01, GEH submitted the results of a spectrum of break sizes for the MSLB with a failure of one DPV. GEH provided results for break sizes that are 100, 80, 60, 40, 20, and 10 percent of the double-ended break size. GEH demonstrated that, for each of these break sizes, the “minimum chimney static head level above vessel zero” remains above the TAF. In RAI 6.3-10 S02, the staff asked GEH to clarify the language “minimum chimney static head level above vessel zero.” In the DCD, GEH calculates this as the static head in the chimney, added to the elevation of the top of the core. In the DCD, GEH also uses “minimum chimney static above vessel zero” but does not use the qualifying statement that “DCD chimney static head is calculated by adding the static head in the chimney to the elevation of bottom of chimney.” RAI 6.3-10 S02 also requested that GEH clarify whether the level calculation accounts for the void fraction of the core. The staff also noticed that, although the core remains covered for all the break sizes, there is a decreasing trend from 40 percent and 20 percent down to 10 percent. The staff also requested in RAI 6.3-10 S02 that GEH address the break sizes below 10 percent and provide the maximum break size that does not exceed the makeup system. In a relevant RAI 6.3-77, as described at the beginning of Section 6.3.2.3.5 of this report, the staff requested the applicant to explain the calculation method for determining the “Minimum chimney static head level above vessel zero per active single failure m (ft).” GEH explained, in the response to RAI 6.3-77, that “chimney static head level with reference to vessel zero” is calculated by adding the equivalent height of water corresponding to the static head of the two-phase mixture inside the chimney to the elevation (7.896 m [25.91 ft]) of the bottom of the chimney. Furthermore, in its response to RAI 6.3-10 S02, GEH explained that the level calculation did not account for the void fraction of the core. Since the calculation showed that there is a certain amount of collapsed water above the active core, the method of showing that the core is covered by water is acceptable. GEH discussed the relationship of break sizes and the minimum water level and argued, by extrapolating, that the water level still covers the top of the core for smaller break sizes. The staff does not agree with extrapolating results for the smaller sizes. However, GEH further explained that the normal reactor water makeup system is the feedwater system, and its capability is sufficient to provide inventory makeup for an 80-percent MSLB. In reality, small-size breaks would be well within the makeup capability of the feedwater system. In the event a small break occurs that does not cause containment pressurization, the break would be detected by the LD&IS. Considering the ESBWR makeup water capability and the ADS, the staff accepts that there is no need to further analyze break sizes below 10 percent of an MSLB. And, because there is no core uncovery and the containment pressure is within limits, the response for the smaller break sizes is satisfactory. The technical basis requested by the staff in RAI 6.3-16 for the settings of the timer delays associated with the ECCS initiation logic were incorporated into the response of RAI 6.3-10 as described above. Based on the applicant’s responses, RAI 6.3-10 and RAI 6.3-16 are resolved.

In RAI 6.3-76, the staff asked GEH to explain why the bounding steamline break gives a higher collapsed liquid level in the chimney than the nominal case. In response, GEH showed plots comparing the downcomer and collapsed chimney level for the nominal and bounding cases. The collapsed level in the chimney is directly related to the level in the downcomer because of the manometer effect. For the bounding case, the downcomer reaches a lower collapsed level at a later time than for the nominal case. This causes the GDCS injection phase to begin later in the bounding case transient. At this time, the core void fraction will be lower and the decay heat reduced from the nominal case. During the injection phase, the collapsed level in the chimney will experience oscillations resulting from the interaction of the core void with incoming subcooled water from the lower plenum. The lower core void fraction and decay heat will reduce the magnitude of the oscillations. For the nominal case, the minimum static head in the chimney occurs during these oscillations, whereas for the bounding case, it occurs before.

In RAI 6.3-76, the staff requested the applicant to explain the reason for the minimum chimney static head for the steam line break inside the containment for the cases run with bounding values are higher than those run using the nominal values. Although the bounding steamline break inside containment gives a higher collapsed liquid level in the chimney than the nominal case, the staff finds that the analysis is conservative and still shows that the ESBWR design has a safety margin with respect to this event. The applicant adequately explained the differences in timing and magnitude for the interaction between the downcomer level and chimney level, as well as the differences between the bounding cases and the nominal cases, and hence the staff agrees with the explanation given by GEH in its response to RAI 6.3-76. The minimum collapsed chimney level for the nominal case happens during the GDCS injection phase, when the core is experiencing oscillations in level. The minimum collapsed chimney level for the bounding case happens just before the injection and before the core oscillations. The staff reviewed the plots submitted by GEH in response to RAI 6.3-76. The plots show that, for the nominal and bounding conditions, the bounding case is qualitatively a more conservative analysis. Both analyses demonstrate that the ESBWR design has margin to core uncovery for this event, and, therefore, the staff finds the results of the analyses acceptable and RAI 6.3-76 is resolved.

The staff did not request that GEH provide an analysis of the MSLB outside containment. This event is bounded by the MSLB inside containment.

6.3.2.3.6 Evaluation Model Parameters and Assumptions

The following sections discuss the staff's review of the evaluation model parameters and assumptions to ensure that the applicant chose them conservatively.

Initial Power Level DCD Tier 2, Revision 9, Table 6.3-11, states that GEH is using a core power of rated +2 percent for its bounding LOCA analysis. This is consistent with the requirements in SRP Section 15.6.5. The staff finds this value acceptable.

Maximum Linear Heat Generation Rate DCD Tier 2, Revision 9, Table 6.3-11, states that GEH is using a peak linear heat generation rate of 44.95 kW/m (13.7 kW/ft) for its bounding LOCA analysis. This value is consistent with the limit in NEDO-33242, Revision 1, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report," which gives a thermal-mechanical limit of 43.96 kW/m (13.4 kW/ft). For the LOCA event, GEH has shown that the ESBWR will experience no core uncovery. Because of the high margin to safety limits for this set of events, the staff finds that the maximum linear heat generation rate (MLHGR) for the ESBWR will be limited by the fuel rod thermal-mechanical design or minimum critical power ratio; therefore, the staff finds the value used for the LOCA analysis acceptable. The staff understands that the Core Operating Limits Report will specify the MLHGR. In accordance with the requirements of 10 CFR 50.46, GEH will update the LOCA evaluations, in the event that the MLHGR specified in the Core Operating Limits Report is allowed to exceed that used in the current LOCA analyses of record.

Axial Power Shapes In RAI 6.3-50, the staff requested the applicant to provide the axial power shape used to perform the nominal and bounding LOCA analysis and provide a discussion on how this shape was selected. In response, GEH submitted the power shape used for the LOCA analyses. The staff finds that the power shape submitted may not be the most conservative for LOCA applications where the core experiences heatup; however, since the core remains covered during all analyzed LOCA transients, and the limiting bundle does not heat up, other power shapes would not produce appreciably different results. Therefore, RAI 6.3-50 is resolved.

Initial Stored Energy For the ESBWR LOCA analyses, GEH used a constant gap conductance. The gap conductance values come from the GEH GSTRM fuel mechanical code. Section 4.2 of this report describes the applicability of the GSTRM code to the ESBWR. Since the LOCA event for the ESBWR does not cause any core heatup, and the core remains covered throughout the entire transient, the staff finds that these values will not have any effect on the calculated figure of merit (i.e., collapsed chimney level) for the LOCA transient and, therefore, finds their use acceptable.

The TRACG04 code uses fuel thermal conductivity values based on the PRIME03 code. As the NRC has not reviewed and approved PRIME03, RAI 6.3-54 asks GEH to justify using this model. RAI 6.3-55 also asks GEH to justify using gap conductance and fuel thermal conductivity from different models. RAI 6.3-54 and RAI 6.3-55 were being tracked as open items in the SER with open items. In response to RAI 6.3-54 S01, the applicant provided evidence, including sensitivity studies, showing that the results analyzed with the GSTRM and

PRIME models are very similar. The staff also performed fuel conductivity sensitivity LOCA confirmatory calculations using the TRACE mode, and the results showed that the minimum water level in the limiting LOCA is not sensitive to the 30-percent conductivity reduction. Therefore, the staff concludes that GEH modeled the initial stored energy properly. Based on the applicant's response, RAI 6.3-54 and RAI 6.3-55 are resolved. Section 21.6.3.2.14 of this report contains a detailed discussion of the evaluation of RAI 6.3-54.

Control Rod Insertion RAI 6.3-52 asks GEH to provide the scram time delay and justify the delay time selected. During a review of TRACG, as applied to the ESBWR LOCA analyses, GEH stated that the travel time of the rods into the core is factored into the decay heat curve. The applicant submitted its response to RAI 6.3-52. The trip delay time of 2.25 seconds is based on 2.00 seconds for the sensor delay, 0.05 seconds for the sensor trip scram solenoid to de-energize (RPS logic), and 0.20 seconds for the scram solenoid de-energize rods to start to move (scram valve open). GEH used a TRACG trip card to model this trip delay time and provided sufficient detail on how it modeled the travel time of the rods. RAI 6.3-52 was being tracked as an open item in the SER with open items. Since the applicant adequately explained the bases for the set point of the timer delay and adequately modeled it the applicant's response is acceptable, RAI 6.3-52 is resolved.

RAI 6.3-80 requested clarification of decay heat selections. RAI 6.3-80 was being tracked as an open item in the SER with open items. In earlier DCD revisions, GEH inconsistently described the decay heat standard used in the TRACG model. In its response to RAI 6.3-80, GEH clarified a typographical error, stating that it based the decay heat curve on the ANSI/ANS 5.1-1994 standard, "Decay Heat Power in Light Water Reactors," and that there were no inconsistencies in ECCS performance analysis in the DCD. Based on the applicant's response, RAI 6.3-80 is resolved.

RAI 6.3-62 requested further details on decay heat modeling. In response, the applicant gave details regarding the power used in the LOCA analysis. RAI 6.3-62 was being tracked as an open item in the SER with open items. The ESBWR decay heat calculations were generated based on the ANSI/ANS 5.1-1994 standard, with additional terms for a more complete shutdown power assessment. The heat sources in the model include decay heat from fission products, actinides, and activation products, as well as fission power from delayed and prompt neutrons immediately after shutdown. The model considered the effect of neutron capture in fission products. GEH assumed end-of-cycle exposure and a conservative irradiation time for decay heat calculations. The irradiation time is the most sensitive input in the decay heat model. Increasing the irradiation time resulted in increased contributions from the long-lived actinides, thus resulting in higher shutdown powers. Since the decay heat is calculated following the appropriate standard ANSI/ANS 5.1, "Decay Heat Power for Light Water Reactors 1994," the staff considers that the assumption used for the decay curve is conservative. In addition, GEH provided assumptions of scram delay times, which included instrument detection of the plant parameters and the delay from signal processing. In the RAI, the staff also asked GEH to justify using the same decay heat curve for both small- and large-break LOCAs. GEH provided a power comparison between the end-of-cycle MSIV closure transient and a decay curve used in the LOCA analysis. The MSIV closure transient experiences a power increase at the beginning, caused by negative void feedback, compared to the power response in a small-break LOCA. The comparison showed that the decay heat curve bounds the MSIV transient power curve, which implies that the decay curve will bound the small-break LOCA as well, and the decay curve used is conservative. However, from the RAI discussion, the staff noticed that the assumptions for the scram signal delay time in the MSIV closure transient differ from those in the LOCA event. The staff estimated additional energy for the small-break LOCA, taking account of the scram delay time. This additional energy could boil off an extra amount of water in the vessel. The staff estimated the extra amount of water and, by subtracting this amount from the GEH minimum level prediction, estimated a new minimum water level. The estimated minimum water level is still above the top of the active core. Considering that other conservative assumptions are in the decay power calculation, the staff accepts the GEH approach of a single decay curve for all LOCA analyses. Based on the applicant's response, RAI 6.3-62 is resolved.

Boric Acid Precipitation DCD Tier 2, Figure 5.1-1, gives a core volume of 96 m^3 (3,390 ft^3) (which does not include the volume in the chimney, separator, and lower plenum). In RAI 6.3-60, the staff requested the applicant to provide the maximum volume of the SLCS inventory that will be injected so the staff can evaluate the possibility of boron precipitation. In response, GEH gave the maximum volume of the SLCS inventory that can be injected into the core. The volume of each of the two SLCS tanks is 8.31 m^3 (293.5 ft^3), giving a total possible SLCS injection inventory of 16.62 m^3 (586.9 ft^3). The SLCS tanks are at ambient temperature, with a 12.5 weight-percent (wt%) sodium pentaborate solution. The volume of the core is more than 5 times that of the SLCS tanks. Since there is no core uncovery and the amount of boron is relatively small unlike in PWRs and will be diluted, the staff finds that it is unlikely that boron will precipitate during a LOCA event in the ESBWR and, therefore, finds that the failure of GEH to analyze this possibility is acceptable and RAI 6.3-60 is resolved.

Containment Pressure Response Section 6.2 of this report discusses the containment pressure response.

ECCS Strainer Performance Evaluation Section 6.2.1.7.3 of this report addresses ECCS strainer performance.

6.3.2.3.7 Reactor Protection System and Emergency Core Cooling System Actions

The staff reviewed the timing, sequencing, and capacity of the RPS and ECCS in relation to the design-basis LOCA analyses. In Revision 6 of the DCD, GEH stated that the ICS drain line break with failure of one GDCS injection valve is the limiting break for the minimum collapsed chimney level for the ESBWR. Section 6.3.2.2.7 of this report describes the sequence of the RPS actions. The sections below discuss the evaluation of the RPS and ECCS functions for the design-basis events presented in DCD Tier 2, Revision 6.

In RAI 6.3-56, the staff asked for more details on the sequence of events for several pipe breaks than were provided in DCD Tier 2, Tables 6.3-7 through 6.3-10. The staff asked GEH to include trip signals and setpoints for all RPS actions, as well as the actions necessary for long-term core cooling. RAI 6.3-56 was being tracked as an open item in the SER with open items. GEH responded that it revised DCD Tier 2, Tables 6.3-7 through 6.3-10, to include the detailed sequence-of-events information and signals for all expected RPS actions. The RPS trip signals included are high drywell pressure and reactor water Level 3. In addition, the subject tables include ECCS initiation signals. The analyses show that no operator actions are required to support long-term core cooling (e.g., opening the GDCS equalizing line valves from the wetwell suppression pool to the RPV) for the timeframe of the sequence-of-events tables. DCD Tier 2, Section 6.3.2.7, describes the actions supporting long-term core cooling beyond the timeframe established in the sequence-of-events tables, if required. This section explained that the long-term portion of GDCS can begin operation following a longer equalization valve time delay initiated by a confirmed ECCS initiation signal and by the RPV level reaching Level 0.5, which is 1 m (3.28 ft) above the TAF. The response to RAI 6.3-56 provided sufficient information on the sequence of events and trip signals for RPS actions. Based on the applicant's response, RAI 6.3-56 is resolved.

Reactor Scram For a LOCA event, the mitigation function of the reactor scram is to shut down the nuclear chain reaction and reduce power to decay heat levels. For the design analyses, the reactor scram signal is from the loss of power generation buses (i.e., a LOOP that results in a loss of all feedwater). DCD Tier 2, Revision 9, Section 7.2.1.2.4.2, gives a complete list of reactor scram signals. Those that would likely cause the reactor to scram during a LOCA include the following:

- High drywell pressure
- Loss of power generation buses
- Reactor water level reaching Level 3 and indicating that it is decreasing
- MSIV closure indication
- Manual

The staff finds that the timing and function of the reactor scram are adequate for it to perform its safety function.

Isolation Condenser System The LOCA mitigation function of the ICS is to provide injection under high-pressure conditions from the drain lines. In addition, the IC will be used to condense the RPV steam. The IC drain line valves open on the same signal that scrams the reactor. This occurs upon the loss of power generation buses (i.e., a LOOP that results in a loss of all feedwater). DCD Tier 2, Revision 9, Section 7.4.4.3, gives a complete list of IC actuation signals. The following signals would likely cause the IC to actuate during a LOCA:

- Loss of power generation buses
- Reactor water level reaching Level 2 with a time delay
- Reactor water level reaching Level 1
- Loss of feedwater
- MSIV closure indication
- Manual

The staff finds that the timing and function of the ICS are adequate for it to perform its safety function.

MSIV Closure The MSIV closure helps mitigate the depressurization and loss of inventory during a LOCA. The MSIV closure in the limiting LOCA analysis (ICS drain line break with failure of one GDCS injection valve) is initiated on low MSL pressure (plus a delay). The MSIV will also close, based on a Level 2 signal plus a 30-second delay. The staff finds that the MSIV closure is adequate to perform its mitigation function during a LOCA.

ADS Actuation The purpose of the ADS is to depressurize the reactor vessel so that the GDCS can inject cooling water. The ADS is initiated upon confirmation of the Level 1 setpoint or drywell high pressure. Confirmation of Level 1 occurs when it persists for 10 seconds, and confirmation of high drywell pressure occurs when it persists for 60 minutes. Section 6.3.2 of this report discusses the ADS, including the sequencing of the valves. The results of the ECCS performance analyses show that the ADS initiation, sequencing, and capacity enable it to perform its ECCS safety function.

SLCS Actuation The LOCA mitigation function of the SLCS is to provide additional injection inventory under high-pressure conditions. The SLCS timer is initiated upon confirmation of the Level 1 setpoint. The SLCS will actuate after a 50-second delay. The results of the ECCS performance analyses show that the SLCS initiation and capacity enable it to perform its ECCS safety function.

GDCS Actuation The main function of the GDCS is to provide low-pressure coolant inventory in the event of a LOCA, once the RPV is depressurized. The GDCS timer is initiated upon confirmation of the Level 1 setpoint or a sustained high drywell pressure. The GDCS squib valves will then actuate after a 150-second delay. GDCS pools will drain, once the RPV depressurizes below that of the GDCS. During the later stages of the GDCS injection phase of the LOCA, the collapsed chimney level experiences large oscillations. In NEDC-33083P-A, GEH stated that these are manometric oscillations. These oscillations occur as the voids in the core are quenched and a larger static head is created inside the shroud that reduces the flow from the downcomer, leading to an increase in void fraction. The increase in void fraction will cause a decrease in static head inside the shroud, and the downcomer flow will increase and quench the voids, to start the cycle all over again. This is also why the downcomer shows oscillations. Since the channel represented in the ECCS performance plots of collapsed chimney level is the hot channel, the oscillations shown in the chimney are much larger. The staff believes that this may also be a result of geysering. In either case, the staff does not find these observed oscillations to be a safety concern. The mechanism for these oscillations requires that there be recirculation flow and water above the TAF. In addition, at decay heat levels, the core would need to experience a sustained uncovery to heat up to levels that would cause fuel damage. These oscillations currently do not show that the level goes below the TAF. Overall, the results of the ECCS performance analyses show that the GDCS is capable of performing its ECCS safety function.

6.3.2.3.8 Long-Term Core Cooling

In a letter dated October 6, 2005 (See ADAMS Accession Number ML053140221), GEH provided the long-term core cooling calculations. These calculations show that the core remains covered for up to 12 hours. The calculations do not show the levels up to 72 hours. The staff requested this information in RAI 6.3-64 and RAI 21.6-98. RAI 6.3-64 and RAI 21.6-98 were being tracked as open items in the SER with open items. In response to RAI 6.3-64 S01, the applicant updated these calculations to reflect the most recent design. The original RAI response provided the limiting case for the containment LOCA, which is a GDCS line break with one DPV failure. RAI 6.3-64 S01 asked why GEH did not choose a vessel-level limiting case for the long-term safety analysis. GEH provided plots in the supplement for the limiting case and explained that the level response in the short term is more important. The long-term calculation showed that the core is covered with water. GEH further stated that it originally included the discussion of the treatment of noncondensable gases in the analysis coverage in its response to RAI 21.6-96, and it would clarify it further in the pending response to RAI 21.6-96 S01. GEH also agreed to include a discussion of the GDCS bounding case in DCD Tier 2 Section 6.3.3.7.9. The staff confirmed the inclusion of the GDCS bounding analysis in Revision 5 to the DCD. The staff concurs that the minimum water level is determined in the short term, after the break initiation, and agrees that the containment wall condensation has no major impact on the equilibrium RPV level and that the long-term level in the vessel will be filled up to the break location or spillover hole. The response to RAI 6.3-64 S01 provided analysis results showing that the reactor core is covered by water up to 72 hours. Based on the applicant's response, RAI 6.3-64 is resolved. The staff documented its evaluation of the response to RAI 21.6-98 in Section 21.6 of this report.

In RAI 6.3-45, the staff asked GEH to explain the differences between the TRACG input decks used to calculate minimum water levels and those used to perform the containment peak pressure analyses. RAI 6.3-45 was being tracked as an open item in the SER with open items. In RAI 6.3-45 S01, the staff asked GEH to justify its assertion that, even though the input deck for calculating minimum water levels lacks the modifications applied to the containment input deck, the results are still accurate and conservative for the long-term core cooling analysis. GEH responded to RAI 6.3-45 S01 that the analyses in DCD Tier 2, Revision 4, had reconciled the model differences described in its original response to RAI 6.3-45. GEH used a consistent set of assumptions, the same TRACG model, and a consistent input deck to calculate minimum water levels and to perform containment peak pressure of nominal cases. However, the assumptions made for the bounding cases for the containment analysis and the RPV water level analysis were different. GEH updated the table in its response to RAI 6.3-45 and explained the differences for the bounding cases; these differences include the normal water level in the downcomer and suppression pool. The staff agrees with GEH that using the lower water level in the minimum water level calculation is bounding for the LOCA analysis and using a higher water level in the suppression pool is bounding for the peak containment pressure calculation. GEH clarified the difference between the minimum water level calculation and the peak containment pressure analyses. Based on the applicant's response, RAI 6.3-45 is resolved.

6.3.2.3.9 Loss-of-Coolant-Accident Analysis under Feedwater Temperature Operating Domain

GEH submitted NEDO-33338, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," Revision 0, in October 2007. For plant operation with feedwater temperature maneuvering (increase and reduction), GEH evaluated the GDCS injection line breaks for the initial core at the increased and decreased feedwater temperature operating points. The applicant did not find any significant chimney-level differences in the LOCA analysis among the core performance setpoints at SP0, SP1, or SP2. However, in the LTR, GEH did not show the limiting IC drain line break analysis for the expanded operating domain. RAI 6.3-82 asked GEH to provide an analysis for limiting break cases in the high and low feedwater temperature operating points. In response to RAI 6.3-82, GEH committed to providing the limiting break in the high and low feedwater temperature operating points. The staff verified that GEH analyzed the limiting break at the requested operation points, SP1 and SP2, and demonstrated that the minimum chimney water levels are above the TAF in NEDO-33338, Revision 1. The staff concludes that the ESBWR LOCA analysis showed that the reactor can safely operate in the expanded feedwater temperature operating domain, and RAI 6.3-82 is resolved.

6.3.2.3.10 Independent Staff Calculations

Plant Model

The staff used the TRACE thermal-hydraulics code model and independently verified the ESBWR system response in the event of a LOCA. The staff based its confirmatory calculations on the ESBWR design documented in Revision 5 of the DCD. The breaks examined were the MSLB, the FWLB, the IC line break (ICLB), the GDCS line break (GDLB), and the bottom drain line break (BDLB). The staff made the calculations with and without an IC heat transfer (ICHT) to investigate the GEH assumption of no ICHT in its safety analyses. The heat structures connecting the IC to the pool were removed for the calculations without ICHT. The water inventory of the IC is kept available to the RPV. In addition, the staff performed a fuel conductivity sensitivity study to examine how sensitive the minimum water level is to the stored energy.

Summary of Results

The staff's study found that the analyzed cases do not show a core uncovery or heatup. A significant difference is seen in the pressure and level response between the cases with and without ICHT. Two effects were observed when the ICHT was removed from the calculation. The first and obvious effect was that removing the heat exchangers reduced the amount of heat removal from the system. A second effect is that more water from the IC drain tanks enters the system in a short time without ICHT, since it is a constant volume draining process and condensation of the steam in the ICs limits the amount of water that can drain into the RPV. A summary of the results is given in Table 6.3-5 in this report. The GDLB is the limiting break for the cases with no ICHT.

The minimum collapsed chimney level was 2.4 m (7.9 ft) above the top of the active core. Applying the additional conservative assumption of maintaining atmospheric pressure in the wetwell gas space lowers the minimum chimney level to 2 m (6.6 ft). The selection of a limiting GDLB agrees with Revision 4 of the DCD but does not agree with Revision 5. Revision 5 of the DCD shows that the limiting LOCA is an ICLB. The reactor responses in ESBWR LOCAs have similar characteristics, as the ADS turns the LOCA into a situation similar to a large-break LOCA. The minimum water level prediction is sensitive to the timing of the ADS initiation signal. The level oscillation changed the timing of ADS initiations, which is why a minor parameter change can cause the limiting LOCA to change from one case to another. Finally, the fuel conductivity sensitivity study showed that, with a decrease of 30 percent in the conductivity value, the change in the minimum water level is minimal. The staff calculations confirmed that there is enough water inventory to cover the core in all LOCAs.

Table 6.3-5. Minimum Average Chimney Collapsed Level.

Break	Minimum Level Base	Minimum Level No ICHT
MSLB	3.4	3.6
FWLB	3.0	3.1
ICLB	3.3	3.1
GDLB	3.1	2.4
GDLB Atmospheric wetwell (WW)		2.0
BDLB	3.3	2.7
0.5* BDLB	3.3	2.9
0.25* BDLB	3.6	

6.3.2.4 *Conclusions*

The staff reviewed DCD Tier 2, Revision 9, Section 6.3, and other relevant material regarding the ESBWR ECCS design, including process diagrams. The staff reviewed the ESBWR design bases and design criteria for the ECCS, as well as the manner in which the ESF design conform to the criteria and bases. The staff concludes that the ESBWR ECCS design meets the guidelines of SRP Section 6.3 and the requirements of the following GDC:

- GDC 2, the ECCS is designed to meet the seismic Category 1 requirements and remain functional following a safe-shutdown earthquake (SSE).
- GDC 4, the ECCS design incorporates features that preclude water hammer and excessive dynamic loads.
- GDC 5, the ECCS is designed for a single nuclear power plant and is not shared between units.
- GDC 17, the ECCS performs its functions without relying on onsite or offsite ac power.
- GDC 27 and GDC 35 safety analyses of the design-basis transients and accidents were performed with the assumption that the most reactive control rod stuck out of the core, and the results demonstrate that the ECCS can provide abundant core cooling, so that (1) fuel and clad damage will not interfere with continued effective core cooling, and (2) the acceptance criteria specified in 10 CFR 50.46 for LOCAs are met.
- GDC 36 and GDC 37 the ECCSs and components are designed to permit periodic inspection and testing of the operability of the system throughout the life of the plant.

The ESBWR design includes preoperational testing for the ECCS, as discussed in DCD Tier 2, Revision 9, Section 14.2.8. In addition, DCD Tier 1, Revision 9, Sections 2.1.2, 2.2.4, 2.4.1, and 2.4.2, specify (1) the design commitments of the ECCS, (2) the inspections, tests, or analyses to be performed by the COL applicants, and (3) the acceptance criteria to ensure that the COL applicants build the ECCS as designed. Therefore, the staff finds the ESBWR ECCS design acceptable.

Based on the TRACG analysis provided in the DCD and in its responses to RAIs, GEH demonstrated that there is no core uncovery or heatup for any design-basis LOCA. The fuel does not heat up during a LOCA; therefore, the PCT is expected to be within the acceptance criterion of 1,204 degrees C (2,200 degrees F). There is no additional oxidation of the cladding as a result of a LOCA. There is no additional hydrogen generated from the chemical reaction of the cladding with water or steam, because the temperatures are not high enough to create this chemical reaction. There are no changes in core geometry resulting from a LOCA that would prevent the core from being amenable to cooling. The ECCS conforms to the review guidelines and acceptance criteria of SRP Section 6.3. The staff concludes that the ECCS meets the acceptance criteria of 10 CFR 50.46 and the pertinent requirements of GDC 2, 4, 5, 17, 27, 35, 36, and 37.

6.4 Control Room Habitability Systems

The control room habitability area (CRHA) is served by a combination of individual systems that collectively provide the habitability functions. These systems are the CRHA HVAC subsystem

(CRHAVS), the radiation monitoring subsystem (RMS), the lighting system, and the FPS. The ESBWR design includes features to ensure that the control room operators can remain in the control room and take actions both to safely operate the plant under normal conditions and to maintain it in a safe condition under accident conditions. These habitability features include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation, lighting, personnel and administrative support, and fire protection.

6.4.1 Regulatory Criteria

The staff reviewed the ESBWR DCD Tier 2, Revision 9, Section 6.4, in accordance with SRP Section 6.4, Revision 3, March 2007, which discusses the control room habitability system. Conformance with the SRP acceptance criteria forms the basis for the staff's evaluation of the CRHA systems. The following regulations and NRC guidance documents apply to these systems:

- GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with postulated accidents

- GDC 5, as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s)

- GDC 19, "Control room," as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection

- 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluations and design provisions to preclude certain control room habitability problems

- TMI Action Plan Item III.D.3.4 (NUREG–0737), regarding protection against the effects of toxic substance releases, either onsite or offsite

- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 1, January 2007

- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," May 2003

- Generic Safety Issue, Item B-36, "Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems"

- Generic Safety Issue, Item B-66, "Control Room Infiltration Measurements"

- Generic Safety Issue 83, "Control Room Habitability (Revision 3)"

- Staff Requirements Memorandum (SRM) and SECY 94-084 as they apply to the use of RTNSS to address uncertainties as a defense-in-depth method

The generic safety issues can be found in NUREG–0933.

6.4.2 Summary of Technical Information

The CRHA is served by a combination of individual systems that collectively provide the habitability functions. The systems that make up the habitability systems include the following:

- CRHAVS
- RMS
- Lighting system
- FPS

When ac power is available, the CRHAVS provides normal and abnormal HVAC service to the CRHA, as described in DCD Tier 2, Revision 3, Section 9.4.1. When ac power is unavailable for an extended time, or if a high level of radioactivity is detected in the CRHA outside air supply duct, the RMS automatically isolates the normal air supply to the CRHA. The habitability requirements are then met by the operation of an emergency filter unit (EFU). The EFUs provide emergency ventilation and pressurization for the CRHA. The CRHA is equipped with a variable orifice relief device to ensure that the amount of air exhausted from the CRHA is equal to that supplied. When ac power is unavailable, the CRHA is passively cooled by the CRHA passive heat sink.

The process RMS provides radiation monitoring of the CRHA environment and outside air intake.

The FPS provides smoke detection and fire damper isolation.

The lighting system provides emergency lighting.

The MCR provides storage capacity for personnel support equipment. Manual hose stations outside the CRHA and portable fire extinguishers provide fire suppression in the CRHA.

The CRHA contains the following features:

- Main control consoles and associated equipment
- Shielding and area radiation monitoring
- Provisions for emergency food, water, storage, and air supply systems
- Kitchen and sanitary facilities
- Provisions for protection from airborne radioactive contaminants

The CRHA is contained inside a seismic Category I structure (the control building [CB]) and is protected from wind and tornado effects, external floods and internal flooding, external and internal missiles, and the dynamic effects associated with the postulated rupture of piping.

The habitability systems maintain the MCR environment suitable for prolonged occupancy for the duration of a postulated accident. In particular, the systems ensure the following:

- The MCR is designed to withstand the effects of an SSE and a design-basis tornado.

- The radiation exposure of MCR personnel for the duration of the postulated limiting faults discussed in Chapter 15 does not exceed the limits set by GDC 19.

- The emergency habitability system maintains the fresh air requirements in American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) Standard 62.1, "Ventilation for Acceptable Air Quality," issued 2007, for up to 21 MCR occupants.

- The habitability systems detect and protect MCR personnel from external fire, smoke, and airborne radioactivity.

- The individual systems that perform a habitability system function are automatically actuated. Radiation detectors and associated control equipment are installed at various plant locations, as necessary, to provide the appropriate operation of the systems.

- The CRHA includes all instrumentation and controls necessary during safe shutdown of the plant and is limited to those areas requiring operator access during and after a DBA.

- CRHA habitability requirements are satisfied without the need for individual breathing apparatus or special protective clothing.

- The CRHA EFUs and associated fans and ductwork; the CRHA envelope structures; and the CRHA heat sink, doors, isolation dampers, and valves, including supporting ductwork and piping, and associated controls are safety-related and seismic Category I.

- Nonsafety-related pipe, ductwork, or other components located in the control room are designed, as necessary, to ensure that they do not adversely affect safety-related components or the plant operators during an SSE.

- The EFU trains are designed with sufficient redundancy to ensure operation under emergency conditions.

- The EFUs are operable during a loss of normal ac power.

- The EFUs operate during an emergency to ensure the safety of the control room operators and the integrity of the control room by maintaining a minimum positive differential pressure inside the CRHA.

- The CRHA envelope is sufficiently leaktight to maintain positive differential pressure with one EFU in operation.

- Electrical power for safety-related equipment, including EFUs, dampers, valves, and associated instrumentation and controls, is supplied from the safety-related uninterruptible power supply. Active safety-related components are redundant, and their power supply is divisionally separated, such that the loss of any two electrical divisions does not render the component function inoperable.

The EFUs are redundant safety-related components that supply filtered air to the CRHA for breathing and pressurization to minimize inleakage. The EFUs and their related components form a safety-related subset of the CRHAVS. Each train consists of an air intake, fan filtration housing, ductwork, and dampers.

The EFU delivery and a variable orifice relief device discharge system are optimized to ensure that there is adequate fresh air delivered and mixed in the CRHA. This is accomplished by using multiple supply registers, which distribute the incoming supply air with the control room air volume, and a remote exhaust to prevent any short cycling. The EFU-delivered supply air is distributed in the MCR area of the CRHA. The EFUs turn over the volume of control room air approximately seven to nine times per day.

This diffusion design (mixing and displacement), in conjunction with convective air currents (caused by heat loads or sinks) and personnel movement, ensures that the occupied zone temperature is within acceptable limits, the buildup of contaminants (e.g., carbon dioxide [CO_2]) is minimal, and the air remains fresh.

The "Occupied Zone" of the MCR region is normally occupied and is generally considered to be between the raised floor and 2 m (6.6 ft) above the floor. Short cycling refers to a poor design condition, where the outside air transits the served space and exhausts to the outside without mixing. This occurs when the outside air inlet and room exhaust are in close proximity. The fresh air for the CRHA is supplied at a high elevation and the exhaust for removing the air is below the floor, so the two are not in close proximity to each other.

Control Room Habitability Area

The CRHA boundary is located on elevation –2000 mm (-6.6 ft) in the CB.

The CRHA envelope includes the following areas:

- Administration Area (Room 3270)
- Reactor Engineer/Shift Technical Advisor Office (Room 3271)
- Shift Supervisor Office (Room 3272)
- Kitchen (Room 3273)
- MCR (Room 3275)
- Restroom A (Room 3201)
- Restroom B (Room 3202)
- MCR Storage Room (Room 3204)
- Electrical Panel Board Room (Room 3205)
- Gallery (Room 3206)
- Auxiliary Equipment Operators Workshop (Room 3207)
- Air-Handling Unit (AHU) Room (Room 3208)

These areas constitute the operation control area, which can be isolated and remain habitable for the duration of a DBA if high radiation conditions exist. Potential sources of danger, such as steamlines, pressurized piping, pressure vessels, CO_2 firefighting containers, and the like, are located outside the CRHA.

Heat Sink

The function of a passive heat sink for the CRHA, which is part of the CRHA emergency habitability system, is to limit the temperature rise inside each room during the 72-hour period following a loss of CRHAVS operation.

The CRHA heat sinks consist of the following: the CRHA outer walls, floor, ceiling, and interior walls and access corridors; adjacent safety-related distributed control and information system (Q-DCIS) and nonsafety-related DCIS (N-DCIS) equipment rooms and electrical chases; and CRHA HVAC equipment rooms and HVAC chases. After the 72-hour period, the EFU maintains the habitability of the CRHA using RTNSS power supplies. The recirculation AHU, with supporting auxiliary cooling units, removes heat to support MCR habitability after 72 hours.

Shielding Design

The design-basis radiological analysis presented in the DCD, Chapter 15, crediting the control room protective features, dictates the shielding requirements for the CRHA. DCD Tier 2, Revision 9, Chapter 15, Section 15.4, contains descriptions of the design-basis LOCA source terms, MCR shielding parameters, and evaluation of doses to MCR personnel.

Component Descriptions

The EFU outside air supply portion of the CRHAVS is safety-related and seismic Category I. Two trains, which are physically and electrically redundant and separated, provide single active failure protection. If one train fails, it is isolated, and the alternate train is automatically initiated. Both trains have 100-percent capacity and are capable of supplying 99-percent credited efficiency filtered air to the CRHA pressure boundary at the required flow rate. The exhaust from the CRHA is through a variable orifice relief device, which is safety-related, and its location is optimized to ensure proper scavenging of the air from the control room in an amount equal to the supply. Backflow prevention through the controlled leak path, the variable orifice relief device, is not required, since the CRHA is at a positive pressure during normal and emergency operation. The EFU design uses a pre-filter, a high-efficiency particulate air (HEPA) filter, a carbon filter, and a post-filter to provide radiological protection for the CRHA outside air supply.

The CRHA pressure boundary includes penetrations, dampers and valves (including the variable orifice relief device), interconnecting duct or piping, and related test connections and manual valves. The isolation dampers and valves are classified as Safety Class 3 and seismic Category I. The dampers and valves have spring return actuators that fail closed on a loss of electrical power. Isolation valves are qualified to provide a leaktight barrier for the CRHA envelope pressure. The boundary isolation function of isolation dampers and valves will be demonstrated by pressure testing the CRHA and by inleakage testing.

Tornado protection dampers are a split wing or an equivalent type, designed to close automatically. The tornado protection dampers are designed to mitigate the effect of a design-basis tornado.

Each access to the MCR has two sets of doors, with a vestibule between them that acts as an air lock.

Leaktightness

The CRHA boundary envelope structures are designed with low-leakage construction. The CRHA is located in an underground portion of the CB. The boundary walls are adjacent to underground fill or underground internal areas of the CB. The construction consists of cast-in-place reinforced concrete walls and slabs to minimize leakage through joints and penetrations.

During normal operation, the CRHA is heated, cooled, ventilated, and pressurized by either of a redundant set of recirculation of AHUs and either of a redundant set of outside air intake fans for ventilation and pressurization purposes. During a radiological event or upon loss of normal ac power, an EFU maintains a positive pressure in the CRHA to minimize the infiltration of airborne contamination. The access doors are designed with self-closing devices, which close and latch the doors automatically.

There are double-door air locks for access and egress during emergencies. Interlocked, double-vestibule doors maintain positive pressure, thereby minimizing infiltration when a door is opened. The CRHA remains habitable during emergency conditions.

Emergency Habitability

The CRHA emergency habitability portion of the CRHAVS is not required to operate during normal conditions, with the exception of the variable orifice relief device. This device is in service to exhaust CRHA air during normal and emergency operation. The normal operation of the CRHAVS maintains the air temperature within a predetermined temperature range. This maintains the CRHA emergency habitability system's passive heat sink at or below a predetermined temperature. The normal portion of the CRHAVS operates during all modes of normal power plant operation, including startup and shutdown.

Operation of the emergency habitability portion of the CRHAVS is automatically initiated by either of the following conditions: a high level of radioactivity in the MCR supply air duct or an extended loss of normal ac power.

Operation can also be initiated by manual actuation. If radiation levels in the MCR supply air duct exceed the high setpoint, the normal outside air intake and restroom exhaust are isolated from the CRHA pressure boundary by the automatic closure of the isolation devices in the system ductwork. At the same time, an EFU begins to deliver filtered air from one of the two unique safety-related outside air intake locations. A constant airflow rate is maintained, and this flow rate is sufficient to pressurize the CRHA boundary to at least 31 pascals (Pa) (⅛-in. w.g.) positive differential pressure with respect to the surroundings. The variable orifice relief device exhausts excess air from the CRHA. This device is a locked-in-place orifice or valve set up to maintain CRHA pressure at the delivered flow. The EFU system airflow rate is also sufficient to supply a fresh air requirement of 10.5 l/s (22 cfm) per person for up to 21 occupants.

Airflow in Emergency Mode

The following mechanisms mix the EFU-supplied inlet air with the general CRHA air:

(1) Supply or inlet registers—The mixing is continuous, as EFU-provided outside air is delivered to the CRHA. Each cfm delivered mixes with the control room air as it exits the supply registers. This is the most common type of space air diffusion, called a mixing system. The supply air is delivered through the air inlet registers, which create an air jet that then mixes the outside air with the room air by entrainment (induction); this helps to reduce the jet velocity and equalize the supply air temperature as it enters the CRHA.

(2) Displacement (ventilation) supply or exhaust—As air is supplied to the CRHA, a similar amount is exhausted from the space. This displaced air is designed to exhaust at a remote location to prevent short cycling and ensure a properly scavenged control room.

(3) Equipment and personnel convective plumes caused by air differential temperature and density—The higher temperature of the air surrounding operating equipment and personnel generates convective air plumes that rise out of the occupied zone, along with any pollutants (e.g., body odors). The rising air is replaced by cooler air from below.

(4) Personnel movement—The airflow requirements are derived from the assumed activity level of the CRHA occupants. This activity generates mixing of the CRHA air.

(5) Molecular dispersion—CO2 and other contaminants are moved across a space by molecular dispersion.

The airflow developed in the ESBWR control room during worst case (outside air temp of 47.2 degrees C [117 degrees F]) accident conditions when the CRHA is isolated and the EFU is in operation with passive cooling is as follows and is illustrated in DCD Tier 2, Figure 6.4-2.

The EFU is operating and provides 220 l/s (466 cfm) of clean outside air into the CRHA. This is delivered to the occupied MCR area, primarily, since this area contains the personnel on duty and houses the active electronic equipment. This supply air exits the ductwork at supply air diffusers (4), which perform the mixing mechanism in (1) above. Depending upon the delivered air temperature, the combined mixed volume either rises or drops. During the worst-case accident conditions, where the outside air is 47.2 degrees C (117 degrees F), modeling shows that this air mixture rises above the ceiling, with a larger quantity of heated air in the MCR; the balance is driven primarily by the convective plumes of the equipment and personnel (mechanism [3] above). The combined air, rising above the ceiling tiles, draws the same quantity of air into the MCR from the area below the raised floor (mechanism [2]). This cooler, slow-moving air gradually spreads over the raised floor and displaces the warmer, stale air toward the ceiling, where it leaves the room. The MCR with the high ceiling becomes thermally stratified (i.e., warmer stale air is concentrated above the occupied zone and cool, fresher air is concentrated in the occupied zone). When the cool air encounters a heat source, such as a person or heat-generating equipment, the air heats up and buoyantly rises out of the occupied zone. The hot air, including CO_2 and body-generated odors, rises because of the air density difference, collects above the suspended ceiling, and spills over into the adjacent rooms. The heat is then released to the cooler walls and concrete. Cooler air in these adjacent rooms drops to the raised floor level and through to the common space below the floor. The discharge flow, 220 l/s (466 cfm), of this air exits the MCR at a remotely opposite location from the EFU supply, to prevent any short cycle of the supply air and ensure a constant turnover of the CRHA air. This air is then drawn into the MCR, and the circuit is complete.

A positive pressure is maintained in the CRHA. There is no buildup of CO_2, since these areas are scavenged continuously by the EFU supply and the exhaust airflow of 220 l/s (466 cfm). The exhaust is located in the lower common area of one of the adjacent rooms and is remote from the EFU supply.

With a source of ac power available, the EFU can operate and is controlled indefinitely through Q-DCIS. In the event that normal ac power is not available, a safety-related battery power supply is sized to provide the required power to the EFU fan for 72 hours of operation. The CRHA isolation dampers fail closed on a loss of normal ac power or instrument air.

One of two ancillary diesel generators provides backup power to the safety-related EFU fans (post-72 hours), if normal ac power is not available. These generators support operation of the control room EFU beyond 72 hours after an accident. For a period between 7 days and the duration of the DBA, the safety-related function of the EFU can be powered by offsite power, by an onsite diesel-generator-powered plant investment protection bus, or by continued use of the ancillary diesel generators. DCD Tier 2, Appendix 19A describes the RTNSS requirements for the ancillary generators.

Upon a loss of normal ac power, the initial temperature in the CRHA ranges from 21.1 to 23.3 degrees C (70 to 74 degrees F), and the relative humidity ranges from 25 to 60 percent.

The CRHA temperature and humidity values calculated during the 72 hours following a DBA equal less than 32.2 degrees C (90 degrees F) wet bulb globe temperature (WBGT) index. The 32.2 degrees C (90 degrees F) WBGT index value is the acceptability limit for minimizing performance decrements and potential harm and preserving the well-being and effectiveness of the control room staff for an unlimited duration.

During the first 2 hours of loss of normal ac power, most of the equipment in the MCR remains powered by the same nonsafety-related battery supply that powers the nonsafety MCR equipment. Any time during a loss of normal ac power, once either ancillary diesel is available, it can maintain the environmental conditions indefinitely. This is accomplished through the continued operation of a CRHA recirculation AHU and the auxiliary cooling unit supplied with each recirculation AHU. If this cooling function is lost, the N-DCIS components in the MCR are automatically de-energized. This is accomplished through safety-related temperature sensors with two-out-of-four logic that automatically trips the power to selected N-DCIS components in the MCR, thus removing the heat load caused by these sources. The remaining CRHA equipment heat loads are dissipated passively to the CRHA heat sinks. The CRHA heat sinks limit the temperature rise by passively conducting heat into the concrete thermal mass.

System Safety Evaluation

Doses to MCR personnel are calculated for the accident scenarios where the EFU provides filtered air to pressurize the CRHA. Doses are calculated for the following accidents:

- 1000 Fuel Rod Failure Dose Results, Table 15.3-16
- Radwaste System Failure Accident Dose Results, Table 15.3-19
- LOCA Inside Containment Analysis Total Effective Dose Equivalent (TEDE) Results, Table 15.4-9
- Main Steam line Break Accident Analysis Results, Table 15.4-13
- Feedwater Line Break Analysis Results, Table 15.4-16
- Small Line Carrying Coolant Outside Containment Break Accident Results, Table 15.4-19
- RWCU/SDC Line Break Accident Results, Table 15.4-23

For all events, the control room dose is within the dose acceptance limit of 50 millisieverts (mSv) (5.0 roentgen equivalent man [rem]) TEDE. Chapter 15 contains the details of the analytical assumptions for modeling the doses to the MCR personnel. No radioactive material storage areas are located adjacent to the MCR pressure boundary. The control room ventilation inlet distances from potential release points are maximized to the extent possible. However, the separation distances in SRP Section 6.4 are not always met. Failure to meet these distances is acceptable because the dose analyses developed for the CRHA used the actual plant layout of the CB intake louvers and potential release points.

As discussed and evaluated in SRP Section 9.5.1, the use of noncombustible construction and heat- and flame-resistant materials throughout the plant reduces the likelihood of fire and its consequential impact on the MCR atmosphere. SRP Section 9.4.1 discusses the operation of the CRHAVS in the event of a fire. The exhaust stacks of the onsite standby power diesel generators and ancillary diesel generators are located more than 48 m (157 ft) from the fresh air intakes of the MCR.

The fuel oil storage tanks for the onsite standby power system and the ancillary diesel generators are located more than 55 m (180 ft) feet from the MCR fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the MCR.

DCD Tier 2, Table 6.4-2 lists the typical sources of onsite chemicals, and DCD Tier 2, Figure 1.1-1 shows their locations. The staff analyzed these sources in accordance with RG 1.78, and the methodology in NUREG–0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release," is to be applied on a site-specific basis (Section 6.4.9).

During emergency operations, the design of the passive heat sink for the CRHA emergency habitability system limits the temperature inside the CRHA to 33.9 degrees C (93 degrees F). This maintains the CRHA within the limits for reliable human performance (DCD Tier 2, Revision 9, Section 6.4.10, References 6.4-1 and 6.4-2) over 72 hours. The walls and ceiling that act as the passive heat sink contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours.

DCD Tier 2, Table 3H14 lists the input parameters assumed in the CB heatup analyses. The EFU portion of the CRHAVS provides 220 l/s (466 cfm) of ventilation air to the MCR and is sufficient to pressurize the control room to at least a positive 31 Pa (⅛ inch w.g.) differential pressure with respect to the adjacent areas. This flow rate also supplies the recommended fresh air supply of 10.5 l/s (22 cfm) per person for a maximum occupancy of 21 persons (DCD Tier 2, Revision 9, Section 6.4.10, Reference 6.4-4).

The normal and emergency (i.e., EFU) outside air intake flows are adjusted as required to maintain a minimum flow and, in conjunction with a controlled leak path, maintain a 31 Pa (⅛ inch w.g.) minimum positive pressure in the CRHA, relative to adjacent areas. Flow instrumentation is provided for the fans and AHUs to ensure airflow is maintained above the minimum required.

A low-airflow alarm is provided. CRHAVS differential pressure transmitters are provided to monitor CRHA pressure with respect to adjacent areas and to ensure the pressure is maintained above the minimum positive pressure. A low CRHA differential pressure alarm is provided. A variable leakage device is located under the raised floor to facilitate air circulation and mixing, with sufficient adjustment to maintain the required airflow and CRHA positive pressure, relative to adjacent areas, under all normal and emergency conditions requiring operation of the CRHA AHU or EFU. The CRHA air intake flows and the positive CRHA differential pressure are periodically monitored during operation of the CRHA AHU or EFU.

The airborne fission product source term in the reactor containment following the postulated LOCA is assumed to leak from the containment. The concentration of radioactivity is evaluated as a function of the fission product decay constants, the containment leak rate, and the meteorological conditions assumed. The assessment of the amount of radioactivity within the

CRHA takes into consideration the radiological decay of fission products and the infiltration and exfiltration rates to and from the CRHA pressure boundary. DCD Tier 2, Chapter 15 fully describes the specific radiological protection assumptions used in the generation of post-LOCA radiation source terms.

The use of noncombustible construction and heat- and flame-resistant materials, wherever possible throughout the plant, minimizes the likelihood of fire and the consequential fouling of the control room atmosphere by smoke or noxious vapor. In the smoke-removal mode, a dedicated fan, intake, and exhaust path purge the control room with a high volume of outside air.

The EFU automatically starts during a radiological event, independent of the loss of normal ac power. Through the use of redundant EFU components and dampers, one EFU and supply path to the CRHA would be available during a loss of normal ac power, with failure of up to two divisions of safety-related power, to provide CRHA breathing air and pressurization during a loss of ac power, concurrent with a radiological event. Local, audible alarms warn the operators to shut the self-closing doors, if, for some reason, they are open.

Testing and Inspection

A program of preoperational and post-operational testing requirements is implemented to confirm initial and continued system capability. The CRHAVS is tested and inspected at appropriate intervals consistent with plant technical specifications. Emphasis is placed on tests and inspections of the safety-related portions of the habitability systems. Design changes to the CRHA will ensure key design assumptions are met such as:

- Heat sink / Heat source assumptions
- Air flow assumptions
- Heat transfer values

This will ensure that CRHA calculations and methodologies are maintained and updated throughout the life of the plant.

The applicant provided the following two COL information items:

> <u>6.4-1-A Control Room Habitability Area (CRHA) Procedures and Training</u>
>
> The COL Applicant will verify procedures and training for control room habitability address the applicable aspects of NRC Generic Letter 2003-01 and are consistent with the intent of Generic Issue 83, A Prioritization of Generic Safety Issues, NUREG–0933, October 2006. (ESBWR DCD Tier 2, Reference 6.4-3), System Operation Procedures (ESBWR DCD Tier 2, Subsection 6.4.4), including statements under Testing and Inspection (ESBWR DCD Tier 2, Subsection 6.4.7).
>
> <u>6.4-2-A Toxic Gas Analysis</u>
>
> The COL applicant will identify potential site-specific toxic or hazardous materials that may affect control room habitability to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19. The COL applicant will determine the protective measures to be instituted to ensure adequate protection for control room operators, as recommended in RG 1.78. These protective measures include

features to (1) provide the capability to detect releases of toxic or hazardous materials, (2) isolate the control room if there is a release, (3) make the control room sufficiently leaktight, and (4) provide equipment and procedures for ensuring the use of breathing apparatus by the control room operators.

Testing and Inspection

A program of preoperational and post-operational testing requirements will confirm initial and continued system capability. The CRHAVS is tested and inspected at appropriate intervals, consistent with plant TS. Emphasis is placed on tests and inspections of the safety-related portions of the habitability systems.

Design changes to the CRHA will ensure key design assumptions are met, such as the following:

- Heat sink and heat source assumptions
- Airflow assumptions
- Heat transfer values

This will ensure that CRHA calculations and methodologies are maintained and updated throughout the life of the plant.

Preoperational Inspection and Testing

Preoperational testing of the CRHAVS will verify that the minimum airflow rate of 220 l/s (466 cfm) is sufficient to maintain pressurization of the MCR envelope of at least 31 Pa (⅛ in. w.g.) with respect to the adjacent areas. The variable orifice relief device is set during this evolution to ensure that an equal amount of air is exhausted from the CRHA. The differential pressure transmitters monitor and confirm the positive pressure within the MCR.

The installed flow meters are used to verify the system flow rates. The pressurization of the control room limits the ingress of radioactivity to maintain operator dose limits below regulatory limits. Air quality within the CRHA environment is certified as within the guidelines of ASHRAE Standard 62.1- 2007 requirements for continued occupancy, by meeting the fresh air supply requirement of 10.5 l/s (22 cfm) per person for the type of occupancy expected in the CRHA. The capacity of the safety-related battery is verified to ensure it can power an EFU fan for a minimum of 72 hours. Heat loads within the CRHA are certified as less than the specified values. Preoperational testing of the CRHAVS isolation dampers verifies the leaktightness of the dampers. Preoperational testing for CRHA inleakage during EFU operation is conducted in accordance with ASTM E741, “Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution.” SRP Section 11.5 discusses the testing and inspection of radiation monitors, while Chapter 14 discusses the other tests noted above.

Inservice Testing

Inservice testing of the CRHAVS includes operational testing of the EFU fans and filter unit combinations, EFU filter performance testing, automatic actuation testing of the CRHA isolation dampers and EFU fans, and unfiltered air inleakage testing of the CRHA envelope boundary. The CRHA boundary is pressure tested periodically to verify leaktightness on the envelope walls, doors, and boundaries. The integrity of the CRHA envelope is tested in accordance with RG 1.197 and ASTM E741.

The control room EFU supplies air with a design flow rate of 220 l/s (466 cfm), and it is designed to maintain the control room envelope at a positive pressure, with respect to adjacent compartments, during normal operation and radiological events. An intake filter efficiency of 99 percent is assumed for particulate, elemental, and organic iodine species. The system does not include filtered recirculation, and the design incorporates leaktightness requirements (SRP Section 6.4.3). Although the control room is maintained at a positive pressure, the dose analysis assumes an unfiltered inleakage rate of 5.66 l/s (12.0 cfm).

Based on the ESBWR CRHA design and ventilation system operation, the acceptance criteria for inleakage associated with the CRHA will be no greater than the amount of unfiltered leakage assumed in the dose consequence analysis minus 2.36 l/s (5 cfm), which is the amount of unfiltered inleakage allocated for ingress and egress.

Nuclear Air Filtration Unit Testing

The EFU filtration components are periodically tested in accordance with ASME AG-1-2003, "Code on Nuclear Air and Gas Treatment," to meet the requirements of RG 1.52. Periodic surveillance testing of safety-related CRHA isolation dampers and the EFU components are carried out in accordance with IEEE-338-2006, "Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems." Safety-related CRHA isolation dampers and the EFU are operational during the plant's normal and abnormal operating modes.

Instrumentation Requirements

The MCR contains alarms for the following CRHA/CRHAVS conditions:

- Low airflow (each EFU fan, recirculation AHU, and outside air intake fan)
- High filter pressure drop (each EFU and normal outside air intake filters)
- High space room temperatures (nonsafety-related temperature detection)
- High room temperature (safety-related temperature detection)
- Low room temperature
- Low recirculation AHU entering air temperature
- Low CRHA differential pressure
- Smoke detected
- High and low humidity in the CRHA
- CRHA airlock doors that are open during an SBO
- High radiation in the CRHA
- High radiation in the outside air intake duct

If the redundant, nonsafety-related CRHAVS cooling is lost, and the CRHA temperature increases, safety-related sensors provide a trip signal through the safety-related system logic and control ESF to de-energize selected nonsafety N-DCIS equipment located in the CRHA. Safety-related sensors monitoring CRHA temperatures provide the logic to trip selected N-DCIS loads in the CRHA. A common alarm is provided to indicate a high CRHA air temperature and a potential high thermal heat sink temperature. Furthermore, this high-temperature alarm setting is set below the N-DCIS trip setpoint. This early detection of rising CRHA and heat sink temperatures allows early operator attention and action before selected N-DCIS loads are tripped in the MCR and ensures operators will take appropriate actions before experiencing temperatures in excess of those assumed in the CRHA heatup calculation. CRHA heat sink temperatures are assumed to be within the specified limit if the average of the air temperatures in the heat sink has been within the specified limit. The temperature response of the materials in the CRHA heat sink area is slower than the response of the average air temperature on increasing temperature (i.e., a loss of normal cooling). If the average of the CRHA air temperatures exceeds the specified limit, restoration of the CRHA heat sinks is verified by an administrative evaluation, considering the length of time and extent of the CRHA heat sink average air temperature excursion outside of limits, or by direct measurement of the temperatures of the structural materials in the CRHA heat sink area.

6.4.3 Staff Evaluation

The staff reviewed the information in DCD Tier 2, Revision 9, Section 6.4, and referenced sections, to determine compliance with the GDC, TMI Action Plan items, and other appropriate regulatory criteria and guidance documents.

GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

The CRHAVS and its components are located in a seismic Category I structure that is protected from tornado, missile, pressure, and flood damage. The EFU portion of the CRHAVS is safety-related and designed to seismic Category I standards.

In RAI 6.4-13, the staff asked the applicant to identify which intakes are protected against tornado damage and to provide an assessment of the impact of a sudden pressure drop resulting from a tornado. RAI 6.4-13 was being tracked as an open item in the SER with open items. In response to RAI 6.4-13, the applicant revised the DCD to state that all CRHA ventilation penetrations for outside air intake and exhaust openings have tornado protection. In addition, the CB ventilation systems outside air intake and return exhaust openings have tornado protection. Because the applicant revised DCD Tier 2, Section 9.4.1.1, Design Bases to include a design requirement that all CRHA ventilation penetrations for outside air intake and exhaust openings are provided with tornado protection, the staff finds that this Tier 2 design requirement provides assurance that the CRHAVS components located on the outside of the seismic Category 1 structure will also be protected from tornado and missile damage Therefore, based on the applicant's response, RAI 6.4-13 is resolved.

The design of nonseismic pipe, ductwork for kitchen and sanitary facilities, and other nonessential components in the CRHA ensures that their failure during an SSE will not adversely affect essential components.

Potential sources of danger, such as pressure vessels and CO_2 firefighting containers, are located outside the CRHA.

There are no high-energy lines in the CB that could affect the CRHA; therefore, the habitability systems are protected against the dynamic effects that may result from possible failures of such lines.

The staff finds that the ESBWR CRHA design complies with GDC 4, in that the essentially underground structure is contained within a seismically qualified Class I building and is protected from the effects of external environmental conditions, such as wind, flooding, pipe whip, and discharging fluids from high-energy piping.

GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The staff review finds that the CRHAVS meets the acceptance criteria of GDC 5. The ESBWR control room habitability design supports a single unit. SSCs important to safety are not shared among nuclear power units. Thus, the design satisfied the GDC 5 requirements.

GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions, without personnel receiving radiation exposures in excess of 50 mSv (5 rem) TEDE for the duration of the accident.

Implicit in GDC 19 is that the environmental conditions (such as temperature, humidity, lighting, air circulation, oxygenation, and atmosphere degradation) will be acceptable for personnel and equipment to function. The ESBWR passive reactor design has limited safety-related battery power sources and passive cooling features. The design relies on reducing electrical loads, including lighting, to a minimum, eliminating air recirculation, eliminating nonessential instrumentation and personnel, and other related heat sources, to control power consumption items in the period of 0–72 hours in which a loss of ac power from active sources is not credited.

The applicant justifies the passive control room's reduced function on the basis that, for the first 72 hours, essentially no operator actions are required, and nonsafety-related instrumentation and equipment can be isolated and shut down. Forced air supply during the period of 0-72 hours is unconditioned air distributed by one of two redundant EFUs to occupied areas of the control room. Both control room recirculation AHUs are shut down, and no forced air is supplied to the kitchen, bathrooms, shift supervisor's office, and other areas deemed to be unoccupied. There are substantial concrete physical barriers between unoccupied and occupied areas, but these barriers have openings in the ceiling plenum and floor plenum spaces of the CRHA. Convective air currents exist at some level, and these currents provide the potential for mixing. CRHA air temperatures and air mixing are interrelated and evaluated in a subsequent section.

The staff agrees that the ESBWR passive design features reduce the requirement for operator action in the first 72 hours following an accident. Consequently, this would permit some reduction in the requirements for control room temperature and humidity during this period. However, the essential habitability function of the control room postaccident must still be satisfied. The staff interprets the postaccident function based on GDC 19 and the guidance in SRP Section 6.4 and NUREG–1242. The principal function is to provide a protected and acceptable environment where operators and others who may be present can monitor and maintain the reactor in a safe stable shutdown condition and take action, if necessary, to respond to any adverse performance of systems and components. The actions may involve planning; communicating with State and Federal officials; interfacing with the NRC; evaluating unexpected performance issues, such as a failed component or system; and taking direct physical actions to ensure public health and safety. The staff evaluated the protective and environmental control features discussed below.

In regard to GDC 19 as it applies to radiation protection, the CRHA is well shielded with its position below grade and its enclosure inside the CB. The two principal sources that affect operator dose in the control room are (1) the radiation that bypasses the filter, because of filter inefficiency in the EFU supply air, and (2) the unfiltered inleakage from all other sources.

In RAI 6.4-11, the staff asked the applicant whether the EFU supply louver location, as shown in DCD Tier 2, Revision 3, Figures 1.2.3.and 1.2.11, is consistent with SRP Section 6.4, Revision 3, Acceptance Criterion 5A; specifically, if the louvers are separated from potential release points by 30.5 m (100 ft) laterally and 15.2 m (50 ft) vertically, and whether the actual minimum distances are based on the dose analyses. RAI 6.4-11 was being tracked as an open item in the SER with open items. In response to RAI 6.4-11, GEH confirmed that the ESBWR does not always meet the SRP guidance for intake vertical and horizontal distances from potential release points; however, the dose analyses used actual plant layout data for the intake louver location and release points. The applicant included this information in DCD Tier 2, Section 6.4.5. The staff reviewed the response and the DCD changes. Because DCD Tier 2, Section 6.4.5 was revised to clarify that the dose analyses developed for the CRHA used the actual plant layout of the Control Building intake louvers and potential release points, the staff finds that the separation of intake louvers from potential release points acceptable. Therefore, based on the applicant's response, RAI 6.4-11 is resolved.

The EFU filters are safety-related and designed and tested with appropriate TS surveillances, in accordance with RG 1.52. Other potential sources of leakage into the CRHA are from people entering or leaving and leakages that could occur through cracks and crevices around penetrations or other locations. These other potential sources of leakage are controlled by pressurization of the CRHA to a positive pressure of 31 Pa (⅛ inch w.g.) and by construction and design to ensure very low leakage. The applicant assumed 5.66 l/s (12 cfm) for this unfiltered inleakage in the DBA analysis.

In RAI 6.4-14, the staff asked the applicant to include additional details of EFU supply and purge duct paths. RAI 6.4-14 was being tracked as an open item in the SER with open items. In response, the applicant proposed revisions to DCD Tier 2, Section 6.4.3 that included these details. The staff reviewed the DCD changes and finds that the details adequately clarify and describe the ductwork external to the CRHA associated with the EFU supply, the normal outside air supply, and the smoke purge pathways. Therefore, based on the applicant's response, RAI 6.4-14 is resolved.

In RAI 14.3-152, the staff asked the applicant to provide an ITAAC to verify that the leaktightness of the CRHA had been achieved by testing, in accordance with the guidance in RG 1.197. RAI 14.3-152 was being tracked as an open item in the SER with open items. The applicant clarified that DCD Tier 1, Revision 4, Table 2.16.2-16 added ITAAC 5.b for confirming that Control Room Habitability Area in-leakage does not exceed the unfiltered in-leakage assumed by control room operator dose analyses. In addition DCD Tier 2, Chapter 16, Technical Specification Section 5.5, "Programs and Manuals," includes a section on CRHA boundary control, in which the applicant commits to periodic CRHA leakage testing, performed in accordance with RG 1.197, to verify that the inleakage would not exceed the value assumed in the design-basis analysis. The staff reviewed the RAI response and the referenced Tier 1 and Tier 2 sections and finds that ITAAC and Technical Specification requirements ensure sufficient verification of the initial and periodic leak tightness of the CRHA. Therefore, based on the applicant's response, RAI 14.3-152 is resolved.

The value assumed in the analysis consists of two parts: the assumed leakage of the CRHA, and the value assumed for access and egress. The assumed access and egress value must be subtracted from the assumed unfiltered inleakage value used in the analysis to obtain the acceptance criteria for CRHA testing.

In RAI 6.4-22, the staff asked the applicant to clarify the DCD to clearly state that the ESBWR COL applicant is required to justify a near-zero value for the CRHA access and egress leakage limit. In RAI 6.4-22 S01, the staff requested that the applicant further clarify in the DCD that the acceptance criteria for CRHA unfiltered inleakage will be no greater than the amount of unfiltered leakage assumed in the dose consequence analysis minus the amount of unfiltered inleakage allocated for CRHA access and egress. The staff requested that the applicant revise DCD Tier 2, Section 6.4.4 to include the value assumed for access and egress for CRHA unfiltered inleakage and to provide a basis for the number assumed, or alternatively, revise the DCD to indicate that this number must be specified and justified by the COL applicant.

In response, the applicant revised DCD Tier 2, Section 6.4.7, “Testing and Inspection, Inservice Testing,” to specify 2.3 l/s (5 cfm) as the amount of unfiltered inleakage allocated for CRHA access and egress. The applicant revised DCD Chapter 16, Section 5.5.12, “Control Room Habitability Area (CRHA) Boundary Program,” to indicate that the quantitative limit of unfiltered air inleakage will be the inleakage flow assumed in the licensing basis analyses of DBA consequences, less the amount designated for ingress and egress. The staff finds the proposed DCD changes acceptable because they conservatively allocate a minimum value of unfiltered leakage that is due to CRHA access and ingress and this value is in accordance with SRP Section 6.4. . The staff confirmed that the applicant had incorporated these changes in DCD Tier 2, Revision 7. Therefore, based on the applicant’s response, RAI 6.4-22 is resolved.

In RAI 14.3-153, the staff requested that the applicant provide an ITAAC to verify that the unfiltered leakage is no greater than the value assumed in the dose analysis in DCD Tier2, Chapter 15. RAI 14.3-153 was being tracked as an open item in the SER with open items. Based on a review of the RAI response and the response to RAI 14.3-152 and RAI 6.4-22, as discussed above, the staff finds the responses acceptable because they confirm that DCD Tier 1, Table 2.16.2-16 ITAAC 5.b exists which ensures that CRHA unfiltered inleakage will not exceed the unfiltered in-leakage assumed by the control room operator dose analyses. Therefore, based on the applicant’s response, RAI 14.3-153 is resolved.

The unfiltered inleakage allocation of 2.3 l/s (5 cfm) is reasonable, because, as stated in the DCD, during a radiological event or upon loss of normal ac power, an EFU maintains a positive pressure in the CRHA to minimize infiltration of airborne contamination. The access doors are designed with self-closing devices, which close and latch the doors automatically. There are double-door air locks for access and egress during emergencies. Interlocked double-vestibule doors maintain the positive pressure, thereby minimizing infiltration when a door is opened.

The staff finds that the test acceptance criterion for CRHA unfiltered inleakage is in accordance with SRP Section 6.4 and RG 1.197 guidance.

It is acceptable to the staff for the applicant to test to the low-leakage criteria, if the assumptions are justified and if the applicant performs the test in accordance with the requirements of RG 1.197. The staff finds that, through control of inleakage from filter inefficiency or other unfiltered sources, and by acceptable results in the dose consequence analyses, the applicant has provided adequate radiation protection for control room operators.

In RAI 6.4-12 and RAI 6.4-15, the staff requested that the applicant identify the design features in the ESBWR standard design that mitigate the consequences of a toxic gas event. RAI 6.4-12 and RAI 6.4-15 were being tracked as open items in the SER with open items. In response, the applicant explained that the ESBWR design does not make specific provisions for toxic gas control. Instead, the ESBWR design identifies COL information items whereby each COL applicant must review the potential effects of toxic gas spills on the specific site, near the site, or in transportation modes in the vicinity of the site, in accordance with RG 1.78. In the event toxic gas levels exceed guidance values in the CRHA, the COL applicant must submit a plan acceptable to the staff that provides for the protection of control room operators. The staff reviewed the RAI responses and COL Information Item 6.4-2-A and finds them acceptable because a COL information item requires a toxic gas review to be performed by an applicant that references the ESBWR standard design. The details of any required design provisions, required by the plan, to mitigate the consequences of a toxic gas event would be provided by a COL applicant. Therefore, based on the applicant's responses, RAI 6.4-12 and RAI 6.4-15 are resolved.

In regard to GDC 19 as it applies to air quality in the MCR, the number of occupants affects the freshness of the air and cooling or heating loads. The ESBWR designed the air supply to provide 220 l/s (466 cfm). A review of ASHRAE 62.1-2007 indicates that this is more than sufficient for the 11 personnel assumed to occupy the CRHA during postaccident isolation. The staff considered the guidance of NUREG–1242 and concluded that, postaccident, there would be an expanded control room occupancy that may include a utility executive, an NRC observer, a communications specialist, five operators, and potentially two individuals from the Technical Support Center staff, if the Center is not available, and that the air supply would be sufficient. In RAI 9.4-57, the staff asked the applicant to describe how the design-basis assumptions on CRHA occupancy will be controlled throughout the life of the plant. In response, the applicant revised DCD Tier 2, Sections 6.4.5 and 6.4.7, to identify critical key assumptions, such as heat sink values, that will be controlled through procedures. The applicant indicated that DCD Tier 2, Section 17.4, ensures that relevant aspects of plant operation are maintained. COL Information Item 6.4.1-A directs COL applicants to develop procedures to control such parameters for the CRHA. The staff finds the response, including the proposed DCD changes, acceptable because COL Information Item 6.4-1-A requires a COL applicant to develop procedures and training for control room habitability that specifically address statements under Testing and Inspection section, DCD Tier 2, Section 6.4.7. DCD Tier 2, Section 6.4.7 states, among other things, that assumption for heat sources will be maintained throughout the life of the plant. The staff confirmed that these changes were incorporated in DCD Tier 2, Revision 7. Based on the applicant's response, RAI 9.4-57 is resolved.

With regard to GDC 19 as it pertains to control room air quality, the staff reviewed provisions for temperature control, air supply distribution, and mixing. For normal operation the staff finds that the ESBWR design provides sufficient conditioned air with adequate recirculation by the nonsafety-related supply fans and the RTNSS-qualified AHUs, with the associated heating and cooling coils. The recirculation AHU also provides humidity control. The system is powered from the station's ac system. The staff also finds that temperature control for postaccident operation is adequate, as long as ac power is available to operate an AHU and the associated heating and cooling equipment.

With regard to postaccident operation, the staff considered a LOCA that included a 0–72 hour operation with LOOP. Alternating current power is not credited from nonsafety-related sources for 72 hours following the accident. The applicant evaluated the impact on control room temperatures for both the 0 percent exceedance summer design condition of 47 degrees C (117 degrees F), with 20 percent relative humidity, and the winter design condition of -40 degrees C (-40 degrees F).

The staff acknowledges that the concurrence of a LOCA with a LOOP at the maximum or minimum design temperatures would be a statistically small occurrence. Also, the redundant ancillary diesel generators are RTNSS-qualified, and their availability is controlled through the Availability Controls Manual and the Maintenance Rule. In addition, it usually takes much less than 72 hours to restore a LOOP in most instances.

In RAI 6.4-7, the staff asked the applicant to describe how the temperature is maintained for the entire 30-day accident period, to clarify the need for active CRHA cooling after 72 hours into the accident, and to identify CRHAVS nonsafety-related systems and power supplies included in the RTNSS. In response, the applicant provided a more detailed description of these issues. The applicant submitted revisions to DCD Tier 2, Sections 6.4.3 and 9.4.1.1 that added details on what Control Room Area Ventilation components are providing CRHA cooling at various times during the 30-day accident period. The changes also clarified the need for post-72 hour active cooling in the control room and how this will be provided.

In RAI 9.4-31 the staff requested the applicant clarify design details of the power source for the EFU mentioned in the RAI 6.4-7 response, which was proposed to be used during the post-72-hour period. In response, the applicant modified the design such that the EFUs rely on ancillary diesel generators, which are RTNSS power supplies. As described in Section 9.4.1 of this report, the staff has reviewed and finds acceptable the RTNSS systems associated with the CRHAVS as a means to provide post-72 hour temperature control for the CRHA.

RAI 6.4-7 was being tracked as an open item in the SER with open items. Because on the applicant's DCD changes clarified the role of various CRHA heat removal structures and systems and clarified what CRHA structures were operating for each phase of the entire 30 day accident period, RAI 6.4-7 and RAI 9.4-31 are resolved.

For the first 72 hours after a DBA with a loss of ac power, the CRHA zone is isolated. The unfiltered supply air system is shut down and isolated by safety-related dampers. One of two EFU fans starts and supplies filtered air to the CRHA at 220 l/s (466 cfm). The operating recirculation AHU is shut down. Power for the system is provided by a safety-related battery system. With the isolation of the recirculation AHU, normal temperature control is lost, and air circulation in the CRHA is driven only by the EFU supply fans and convective currents. Air circulation and supply distribution is important in maintaining a uniform bulk temperature throughout the multiroom CRHA and in ensuring fresh air for operators at any location.

The applicant states that the ESBWR uses a passive heat sink consisting of the walls, ceilings, and floors of the CRHA to maintain the temperature at less than 33.9 degrees C (93 degrees F).

In RAI 6.4-8, the staff asked the applicant how it would ensure the initial passive heat sink temperature. RAI 6.4-8 was being tracked as an open item in the SER with open items.

The applicant established the maximum normal operation temperature in the CRHA at 21.1 degrees C (74 degrees F). The maximum temperature in the CRHA is important, in that it establishes the basis for the initial concrete heat sink temperature used in the passive heat sink analysis. In response, the applicant explained that the heat sink temperature will be controlled by a TS 3.7.2 surveillance performed every 24 hours. The staff reviewed the RAI response and finds it acceptable because the surveillance provides assurance that the actual temperatures of the heat sinks relied upon for passive cooling of the control room will be periodically monitored to ensure that they remain conservative with respect to the assumptions uses for these values in the passive heat sink design basis analysis . Therefore, RAI 6.4-8 is resolved.

The staff reviewed the maximum temperature of 33.9 degrees C (93 degrees F), with respect to environmental qualification requirements of safety-related components, and finds that this temperature is acceptable. The staff reviewed the maximum CRHA temperature value of 33.9 degrees C (93 degrees F), as stated in DCD Tier 2, Table 9.4-1, against the mild environment equipment qualification temperature of 50 degrees C (122 degrees F), as stated in the DCD, Appendix 3H, Table 3H-10. RAI 9.4-34 and RAI 3.11-28 were issued to the applicant to resolve staff questions in this area as described below.

In RAI 9.4-34, the staff asked the applicant to clarify whether the design considers the reduced airflow and locally increased temperature inside electrical cabinets during the period of passive cooling, and whether those temperatures pose a challenge to equipment operation.

In RAI 3.11-28, the staff asked the applicant to provide additional details on how the service temperature of electrical equipment, including computer-based instrumentation and control (I&C) systems, will be determined for the ESBWR. In particular the applicant was asked to provide details on this process for equipment that is planned to be located inside electrical cabinets or panels in the RB and the CB. The applicant was also asked to explain how the detailed design and testing of electrical equipment, including enclosures, would be carried out, so that the key assumptions of environmental bounding temperatures in these areas remain conservative.

In response to the RAIs, the applicant revised DCD Tier 2, Sections 3.11.1.3, 3.11.4.3, and 3.11.3.1, to more fully explain the temperature qualification process.

The applicant revised the DCD Tier 2, Section 3.11.1.3, definition of equipment, to indicate that computer-based I&C equipment is defined by the equipment plus its surrounding enclosure. It revised the DCD Tier 2, Section 3.11.4.3, to indicate that system testing of computer-based I&C equipment within its cabinet or enclosure is preferred.

In DCD Tier 2, Section 3.11.3.1, the applicant states that the EQ equipment in the CRHA is to be tested at temperatures that are 10 degrees C (18 degrees F) higher than the maximum temperature to which the equipment is exposed for the worst-case abnormal operating occurrence, with the equipment at maximum loading. The worst-case operating temperature is given at 50 degrees C (122 degrees F), as stated in the DCD Tier 2, Appendix 3H, Table 3H-10. In addition, DCD Tier 2, Section 3.11.3.2, states that margins will be included in the qualification parameters to account for normal variations in the commercial production of equipment and reasonable errors in defining satisfactory performance, and that the environmental conditions shown in the Appendix 3H tables do not show such margins. The staff noted that, in DCD Tier 2, Section 3.11.3.2, the applicant stated that the program margin would be in accordance with the guidance in IEEE-323-2003, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.” The staff infers that the applicant used the +5 degrees C (+8 degrees F) value, as stated in the document.

Thus, since CRHA EQ equipment is to be tested at 60 degrees C (140 degrees F), there is some confidence that the equipment would not fail if actual local temperatures exceed the calculated maximum average CRHA bulk temperature of 33.9 degrees C (93 degrees F) by several degrees. Based on the margin in the assumed normal operating temperature used in the CB heatup analysis, and the conservatism inherent in the EQ process that establishes the equipment service temperature, the staff finds that local temperatures are not likely to challenge component operability before ac power is restored. The staff concludes that, independent of operator actions or offsite support, the CB ventilation system design maintains satisfactory environmental conditions for equipment to function for the first 72 hours after the onset of an accident that assumes that all ac power is lost for this period. Therefore, based on the applicant's responses, RAI 9.4-34 and RAI 3.11-28 are resolved.

The staff considered the impact on operators working in an elevated-temperature environment. The applicant's passive cooling analysis indicates that the temperature in the CRHA would reach a 30 degrees C (86 degrees F) dry bulb bulk temperature in approximately 12 hours. After 12 hours, the temperature rate of change is much lower, reaching a CRHA bulk temperature of 33.5 degrees C (92.5 degrees F) at 72 hours. Humidity may also increase from moisture contained in the supply air. Based on a review of NRC and industry standards, the staff notes that human performance is most frequently assessed based on the WBGT index.

In RAI 6.4-24, the staff asked the applicant to justify the use of a psychrometric wet bulb temperature as a valid index to assess heat stress in the ESBWR CRHA, or alternatively, to amend the DCD to provide a heat stress acceptance criterion and index that is in accordance with NRC guidance. The staff also asked the applicant to demonstrate that such a criterion can be met for the ESBWR environmental footprint. The staff also asked the applicant to identify the associated ITAAC.

In response to RAI 6.4-24 S01, the applicant revised the DCD to state that the WBGT index would be the design-basis means by which a heat stress acceptance criterion would be measured. The applicant stated that the CRHA is designed such that 32.2 degrees C (90 degrees F) WBGT would not be exceeded at the end of 72 hours of passive cooling. The applicant provided an accompanying CONTAIN 2.0 computer code demonstration and revised DCD Tier 1, Table 2.16.2-4, to include an ITAAC 4iii that requires a COL applicant to demonstrate this, using an analysis updated with as-built design information.

The staff compared the proposed DCD revisions and analysis result to NRC and industry guidance and finds that, although high, the applicant's chosen WBGT index acceptance criterion for heat stress at the end of 72 hours of passive cooling would not require compensatory actions, such as stay times. Therefore the staff concludes that the ESBWR CRHA temperature and humidity at the end of 72 hours of passive cooling is acceptable with regard to human performance. The staff confirmed that these changes were incorporated in DCD Tier 2, Revision 7. Based on the applicant's response, RAI 6.4-24 is resolved.

The staff reviewed the analytical basis for evaluation for temperature in the control room habitability area for the 0-72 hour postaccident period. The applicant submitted a passive cooling analysis (Control Building Environmental Temperature Analysis) as part of the licensing basis that evaluates heat transfer by use of the CONTAIN computer code. The results indicate that the maximum bulk temperature reached in the CRHA during the 0–72-hour period is less than 33.9 degrees C (93 degrees F).

The staff finds that the applicant's use of an analytical approach as a method to demonstrate the passive heat removal mechanism and to show that the CRHA bulk temperature will not exceed design-basis limits is reasonable.[1]

In RAI 6.4-16 and RAI 9.4-32, the staff asked the applicant to discuss the need to provide cooling to nonsafety-related heat loads in the CRHA following an accident. RAI 6.4-16 was being tracked as an open item in the SER with open items. In response to these RAIs, the applicant explained that, as stated in DCD Tier 2, Section 9.4.1.2, CRHA nonsafety-related heat loads are automatically de-energized when the CRHA AHUs are not available during the first 2 hours, and discussed operator actions to isolate the nonsafety-related heat loads. The staff finds the RAI responses acceptable because they clarify an analysis assumption on accident heat load: that nonsafety heat loads in the CRHA will be de-energized during such accidents. RAI 6.4-16 and RAI 9.4-32 are therefore resolved.

In RAI 9.4-33, the staff asked the applicant to provide sufficient information needed for the staff to evaluate the performance of the ESBWR passive cooling features. In response, the applicant provided analysis assumptions for the control room design and outside environmental conditions for a single-node model of the CRHA that demonstrates the mechanism by which heat is removed (i.e., the absorption of heat by thermal mass of concrete).

The staff noted some conservative parameters in the Control Building Environmental Temperature Analysis, such as the assumptions used for the heat transfer to the concrete, the conservative assumption regarding the initial heat sink temperatures, and the margin for assumed heat loads. In order to ensure the as-built CRHA design captured these assumptions, the staff asked the applicant in RAI 9.4-55 to incorporate the Control Building Environmental Temperature Analysis in the DCD and revise the ITAAC to specifically refer to this analysis.

In response to RAI 9.4-55, the applicant submitted Control Building Environmental Temperature Analysis , LTR NEDE-33536P, as Tier 2* information, and revised DCD Tier 1, Table 2.16.2-4, to clearly link ITAAC 4i, 4ii, and 4iii to the submitted LTR. The staff confirmed that the changes were incorporated in DCD Tier 2, Revision 7.

Because the DCD changes associated with the RAI response clearly establish the analysis methodology for the passive heat sinks, and because the applicant has made changes to that methodology and its assumptions subject to staff review, the staff finds this acceptable and RAI 9.4-55 and RAI 9.4.33 are resolved.

The staff has reviewed the results of the applicant's Control Building Environmental Temperature Analysis as a basis for meeting the design requirements for the MCR HVAC systems, as stated in Chapter 9, Section 8.2.2.1, of the Utility Requirements Document, and in SRP Section 9.4.1. The staff reviewed the applicant's calculation and performed confirmatory calculations using the same methodology and input assumptions. The staff obtained similar results.

The Control Building Environmental Temperature Analysis, model relies on EFU fan flow for air circulation. Because the applicant chose to model the CRHA as a single node, the design-basis analysis model does not demonstrate the convective mixing mechanism that would also be expected to occur. In addition, the design-basis model does not illustrate pressure changes in

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1 See Yilmaz, T.P., and Paschal, W.B., "An analytical approach to transient room temperature analysis," Nuclear Technology, 114:135–140.

the CRHA caused by temperature differences between the supply and exhaust air during EFU operation.

In RAI 9.4-29 the staff requested that the applicant provide assumptions used to establish the minimum EFU fan flow rate criterion that is used to ensure adequate fresh air supply to the CRHA. The staff also requested additional information on how mixing of air would occur in the CRHA.

In response, the applicant provided the results of an analysis of a multinode GOTHIC model. The results demonstrated temperature stratification in the CRHA and convective mixing. The applicant included CRHA airflow design details obtained from this analysis, including a description and illustration of the airflow expected in the CRHA occupied zone in DCD Tier 2, Section 6.4. Based on a review of the design of the CRHA air distribution system as described in the DCD, the staff finds that such mixing would occur and would improve the air quality and temperature in the CRHA. The staff considers the DCD design requirements for mixing and distributing the EFU-supplied inlet air sufficient to ensure that the air quality will be within ASHRAE Standard 62.1 guidelines. Therefore, RAI 9.4-29 is resolved.

In RAI 9.4-49, the staff requested that the applicant provide additional information on the applicability of ASHRAE 62.1-2007 to a tightly closed facility, such as the ESBWR MCR, and determine whether there are long-term indoor air quality effects on habitability that need to be addressed. The applicant responded that preoperational testing as described in DCD Tier 2, Section 6.4.7 and surveillances as described in Generic Technical Specifications Section 5.5.13 in DCD, Chapter 16, will verify that the minimum air flow rate to the CRHA will be supplied. The applicant clarified that CO_2 and odors will be removed using the CRHA leakage paths, including the controlled leakage path. The applicant clarified the DCD to include a design requirement for 7 to 9 air changes to take place per day in the CRHA, and added details for air supply and exhaust location in the CRHA. The staff finds the RAI response acceptable because the Tier 2 changes clarify the importance of design features to ensure adequate air supply and quality to the CRHA; therefore RAI 9.4-49 is resolved.

The staff reviewed the means by which the as-built CRHAVS heat sink will be analyzed to ensure that it will passively maintain the temperature in the CRHA within the design basis for the first 72 hours following a DBA. The means of verification of this design commitment is a CB temperature analysis, using the as-built heat sink dimensions, thermal properties, exposed surface area, as-built thermal properties of materials covering parts of the heat sink, and the as-built heat loads to confirm the results of the control room design-basis heatup analysis.

A CB temperature analysis will be used to confirm the control room winter design-basis heatup analysis to demonstrate that the CRHA bulk air temperature will not be below 12.8 degrees C (55 degrees F) on a loss of normal heating for 72 hours, given winter design-basis conditions.

The staff reviewed DCD Tier 1, Table 2.16.2-4, ITAAC, 4i and 4ii, and verified that sufficient ITAAC exist to perform a thermal analysis, with as-built design details, that confirms the results of the MCR design-basis heatup analysis.

The staff has considered some use of conservative assumptions in the applicant’s design-basis heatup model, such as the assumed thermophysical properties of CB concrete, the orientation of the CB for the highest solar radiation, a 15-percent margin in the assumed sensible heat load, an assumed CRHA failure 8 hours before the postulated accident (resulting in increased CRHA air and heat sink temperatures at the start of the analysis), and the applicant’s use of higher heat sink temperatures for walls in contact with the ground than would be expected. Based on the use of these conservatisms, and the staff review of the applicant's model, as previously discussed, the staff finds that the applicant has adequately demonstrated that the CB passive heat sinks would likely limit the CRHA occupied zone bulk temperature to below the design-basis temperature of 33.9 degrees C (93 degrees F) for 72 hours, assuming no ac power sources are available for that period. The staff concludes that this bulk temperature would not significantly affect CRHA operator or equipment performance, and the ITAAC acceptance criteria for the summer maximum CRHA bulk average air temperature of 33.9 degrees C (93 degrees F) are acceptable.

The applicant evaluated the minimum CRHA temperature using ECOSIMPRO software, which its consultant developed and owns. The applicant benchmarked the ECOSIMPRO software against the CONTAIN software for the summer design case. The ECOSIMPRO code also assumes a single node for the CRHA. The ECOSIMPRO results showed a minimum bulk temperature in the CRHA of 16 degrees C (61 degrees F) at 72 hours. Based on a review of the analysis results, the staff concluded that the CB passive heat sinks would likely limit the CRHA occupied zone bulk temperature above this design-basis temperature value for 72 hours, assuming no ac power sources are available for that period. The staff concludes that this bulk temperature would not significantly affect the performance of CRHA operators or equipment, and the ITAAC acceptance criteria for the winter minimum CRHA bulk average air temperature of 12.8 degrees C (55 degrees F) are acceptable.

In summary, the staff concludes that the CONTAIN analysis adequately predicts the CRHA occupied zone's maximum and minimum bulk temperatures within the applicant's acceptance criteria. The CONTAIN analysis adequately demonstrates a mechanism of thermal absorption of heat in the CRHA. Verification of the analysis with as-built design and site environmental parameters provides adequate assurance that assumptions in the analysis remain valid. The applicant's maximum and minimum temperature acceptance criteria are adequate to ensure that the CRHA would have an acceptable environment for personnel and equipment in a postulated accident. Thus, the staff concludes that the passive cooling design and associated acceptance criteria are acceptable.

The staff acknowledges that a certain degree of uncertainty remains concerning the performance of the CB ventilation system's unique passive features and the overall performance of the CRHA heat removal system, because of lack of a proven operational performance history. Although not credited by the applicant or the staff to function before 72 hours, the staff notes that the design and regulatory treatment of the ancillary diesels, as described and reviewed in Section 9.4.1 of this report, make it likely that this nonsafety-related source of ac power will be available for CRHA AHU operation before 72 hours. DCD Tier 2, Section 8.3.1.1 states that the ancillary diesel generator automatically starts upon sensing undervoltage on their respective busses. Based on the review of the functional capability and availability of these systems, the staff notes there is additional defense-in-depth protection in the CB ventilation system design to overcome this inherent uncertainty.

In regard to GDC 19 as it applies to air quality the staff reviewed EFU supply register location and provisions for air distribution. During normal operation or post-72 hour operation, the location of the EFU supply registers is not critical and the RTNSS-qualified AHU fully establishes air mixing. During the postaccident 0 to 72-hour operation with a LOOP, air mixing is important to keep localized temperatures from reaching the extremes and to ensure that fresh air is maintained in the operator breathing zone. The applicant located the EFU supply registers just underneath the false ceiling in the occupied zone of the CRHA. The heat or the cold added by the registers would likely be caught in the convective current updraft and distributed to all areas of the CRHA. The staff concluded that some convective currents are probable and that the supply registers in this location have a beneficial effect on mixing.

After 72 hours, the EFU and AHU can be powered by offsite power sources or by two redundant ancillary diesels that start automatically on a LOOP. The staff reviewed the temperature controls in the post-72-hour period for the duration of the accident. The applicant has made provisions to start one of the two RTNSS-qualified AHUs to increase circulation in the CRHA. In addition, the applicant has arranged for additional CRHA cooling to be connected to the AHU cooling water piping outside the CRHA by a valve arrangement.

In regard to GDC 19 as it applies to habitability, the ESBWR emergency lighting provides a minimum luminance of 10 foot-candles (107.6 lux) at all workstations in the main operation areas. This is consistent with the recommendations for emergency lighting in NUREG–0700, "Human-System Interface Design Review Guidelines," issued May 2002. High-efficiency lighting will be used. The applicant assumed a heat load of 400 watts for the emergency lighting in the passive cooling analysis. Although the staff considered this to be a marginal lighting design, it realized that additional portable battery-operated lighting is readily available and could be used to supplement lighting, if needed, for the 0–72-hour postaccident situation with a LOOP.

In regard to GDC 19 as it applies to air supply, stratification, and mixing, the applicant designed the air supply for both normal and postaccident operation on the basis of ASHRAE 62.1-2007, which uses a combination of requirements for personnel and area to determine fresh air requirements. The applicant established 220 l/s (466 cfm) as the supply air flow rate. For normal operation with an AHU providing recirculation, the staff considers the flow rate to be adequate. For postaccident operation with a LOOP, the AHU is isolated. The air in the CRHA is mixed by convective currents, personnel movement, molecular dispersion, and the EFU supply air, with the EFU supply registers located in the MCR area of the CRHA.

The applicant has included design features to promote mixing. The staff finds that these features would promote mixing and mitigate stratification. The staff finds that the ESBWR designed in compliance with ASHRAE 62.1 air quality standards would limit the buildup of other contaminants, such as CO_2, and provide enough mixing to ensure that the CRHA remains at the bulk temperature calculated in the licensing-basis CONTAIN passive cooling analysis. The staff concludes that there would be some convective flow that would augment EFU flow and that air movement would be sufficient to keep the air mixed for freshness and to prevent the buildup of contaminants.

In regard to GDC 19 as it applies to CRHA pressurization and air discharge control, RAI 6.4-9 was being tracked as an open item in the SER with open items. RAI 6.4-9 requested additional information about the adequacy of the EFU system flow rate to maintain CRHA pressurization. In response, the applicant provided an analysis to demonstrate that the control room makeup flow is sized for leakage from the control room boundary when the control room is pressurized to a positive pressure differential of 31 Pa (⅛ in. w.g.). The applicant revised DCD Tier 2, Section 6.4.3, with the results of the analysis. Based on a review of the RAI response and proposed DCD changes, the staff finds the RAI response acceptable because the applicant submitted an analysis, based on the planned leaktight design features that ensured the feasibility of maintaining the tested differential pressure with the design makeup airflow rate in accordance with Standard Review Plan Section 6.4, Revision 3, acceptance criteria item 3. RAI 6.4-9 is therefore resolved.

The staff noted that the applicant did not model the pressure changes in the CRHA caused by temperature differences between the supply and exhaust air during the passive cooling period with the EFU operating.

In RAI 9.4-30 the staff requested that the applicant clarify if changes in the outside environmental conditions such as air temperature and pressure over the accident period could significantly change the volumetric addition of air to the control room such that manual adjustments to the variable orifice device would be required in order to maintain acceptable CRHA positive pressure and makeup airflow rate.

In response to RAI 9.4-30, the applicant modified the CRHA design to add variable orifice relief to maintain a greater-than-31 Pa (⅛ in. w.g.) positive pressure at the minimum flow rate. DCD Tier 1, ITAAC Table 2.16.2-6, Design Commitment 5a, states that the EFUs maintain the CRHA at the minimum positive pressure with respect to the surrounding areas at the required air addition flow rate. This commitment is verified by an ITAAC test.

The variable orifice relief device is manually adjusted, as needed, to maintain CRHA positive pressure. In response to follow-up questions from the staff under this RAI, the applicant demonstrated that required manual adjustments of the device during a postulated accident would be unlikely and would not likely be a burden on operators. The staff finds the ITAAC acceptable because it provides assurance that the design, when built, will supply the minimum required positive pressure at the minimum required air addition flow rate. The staff reviewed the ESBWR Technical Specification TS 5.5.12, Control Room Habitability Program, paragraph d. This paragraph requires periodic measurement at designated locations, of the CRHA pressure relative to all external areas located at the CRHA boundary while the EFU filter is supplying at least minimum airflow rate. The staff finds that this surveillance requirement provides assurance that the controlled leakage path setting will be monitored and adjusted as required during the life of the plant. As discussed in DCD Tier 2, Section 6.4.8, the existence of alarms for low EFU airflow and low CRHA differential pressure assure that these parameters are continually monitored when an EFU is in operation, and that operators would be alerted in a timely manner if any corrective action is required. Based on review of the applicants discussion of impacts to changes in outside air temperature on EFU volumetric flow, the staff agrees that since the percent change of air specific volume is low for a relatively large outside air temperature swing, any changes in outside air temperatures would not affect the performance of the CRHA positive pressure and EFU flow rate parameters during an accident, Therefore frequent adjustment of the CRHA variable orifice relief device is not anticipated. RAI 9.4-30 is therefore, resolved.

The ESBWR provides a variable orifice relief device to maintain a constant 31 Pa (⅛ in. w.g.) positive pressure in the CRHA, while ensuring that the amount of air exhausted from the CRHA is equal to the amount supplied. This device location is optimized to ensure proper scavenging of air from the control room.

The CRHA has the fresh air supplied at a high elevation and the exhaust removed below the floor, so that the supply and exhaust are not in close proximity to each other. The CRHA has a differential pressure indication for monitoring under normal and emergency operation. A low-airflow alarm is provided. Pressure in the CRHA is monitored, and an alarm is actuated if the pressure falls below the setpoint level. In RAI 6.4-23, the staff asked the applicant to revise the DCD to clarify the function, seismic, and safety classification of the variable orifice relief device. In response, the applicant revised DCD Tier 1, Table 2.16.2-3 and DCD Tier 2, Sections 6.4.2, 6.4.4, 6.4.7, 9.4.11, and 9.4.1.2. The staff finds the proposed DCD changes acceptable because the applicant clarified that the device meets SRP Section 9.4.1 design and inservice testing guidance. The applicant's revision to DCD Tier 1, Table 2.16.2-3 lists the CRHA Variable Orifice Relief Device as safety-related, seismic Category 1. These changes clarified the function, seismic and safety classification of the device. The staff confirmed that these changes were incorporated in DCD Tier 2, Revision 7. Based on the applicant's response, RAI 6.4-23 is resolved.

In regard to GDC 19 as it applies to smoke purge, the applicant provided a system for rapid removal of smoke in the event of a fire inside the CRHA. The system is isolatable with safety-related and tornado-protected dampers. The assumption of a CRHA fire postaccident is not a requirement of the design.

To summarize the GDC 19 review, the ESBWR has incorporated design features that protect operators and equipment from radiation, temperature, humidity, and other environmental conditions, and these features are adequate, considering the low probability of adverse events and the availability of defense-in-depth measures.

The regulation in 10 CFR 50.34(f)(2)(xxviii) requires the applicant to evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions, resulting in an accident source-term release, and to make the necessary design provisions to preclude such problems (TMI Action Plan Item III.D.3.4). The design includes adequate protection from radiation, in compliance with GDC 19. The staff finds that this is acceptable.

TMI Action Plan Item III.D.3.4 requires that control room operators be adequately protected against the effects of the accidental release of toxic and radioactive gases and that the nuclear power plant be safely operated or shut down under DBA conditions (GDC 19 in Appendix A to 10 CFR Part 50).

RAI 6.4-17 asked the applicant to state the following in the DCD, regarding testing the CRHA envelope for integrity: (1) that the test requirements and the testing frequency will be consistent with the guidance of RG 1.197, which establishes an inservice test program, and (2) that the test requirements appear in DCD Tier 2, Chapter 16. RAI 6.4-17 was being tracked as an open item in the SER with open items. In response, the applicant stated that DCD Tier 2, Sections 6.4.7 and 6.4.9 were revised to include this information. The staff finds the RAI response and associated DCD changes acceptable because they clarify that testing to demonstrate the integrity of the Control Room Habitability Area envelope is performed in accordance with RG 1.197 and ASTM E741. This is in compliance with SRP Section 6.4, SRP Acceptance Criteria Item 1.E as it applies to CRHA envelope integrity testing requirements and testing frequency, and RAI 6.4-17 is therefore resolved.

The staff concludes that GEH has met the TMI Action Plan Item III.D.3.4 requirements by adding COL Information Item 6.4-2-A in Revision 4 of the DCD Tier 2, Chapter 6. This requires the COL applicant to identify potential site-specific toxic or hazardous materials that may affect control room habitability to meet the requirements of TMI Action Plan Item III.D.3.4. If high radioactivity is detected in the CRHA outside air supply duct, the CRHA normal air supply is automatically isolated, and the GDC 19 habitability requirements are met by an EFU. The EFUs provide emergency ventilation and pressurization for the CRHA. The staff finds that this is acceptable.

Task Action Plan Item B-36 required the development of design, testing, and maintenance criteria for atmospheric cleanup system air filtration and adsorption units for ESF systems and for normal ventilation systems. GEH meets the requirements of Item B-36 by complying with RG 1.52, for the safety-related EFU system, and RG 1.140, for the nonsafety-related filter systems. RAI 6.4-10 was being tracked as an open item in the SER with open items. RAI 6.4-10 requested that the applicant include a reference in the DCD to ASME AG-1, including all addenda. The applicant included this reference in DCD Tier 2, Table 1.9-22. The staff finds that this is acceptable, and RAI 6.4-10 is resolved.

Task Action Plan Item B-66 addresses the magnitude of the control room air infiltration rate. RG 1.197 provides methods acceptable to the staff for determining air infiltration and is referenced in TS Section 5.5.12 of DCD Tier 2, Chapter 16 on the control room habitability boundary. The staff therefore considers the concern of Item B-66 to be satisfied.

Generic Safety Issue 83, "Control Room Habitability" (Revision 3), addresses deficiencies in the maintenance and testing of ESFs designed to maintain control room habitability (e.g., inadvertent degradation of control room leaktightness, shortage of personnel knowledgeable about nuclear HVAC systems). It recommends increased training of NRC and licensee personnel in inspection and testing of control room habitability systems.

GL 2003-01 reemphasized this concern. GEH developed COL Information Item 6.4-1-A ("Control Room Habitability Area (CRHA) Procedures and Training"), which requires the ESBWR COL applicant to verify procedures and training for control room habitability. GEH also added the CRHA Boundary Program (Section 5.5.12) in the DCD Tier 2, Chapter 16 to establish the CRHA boundary test method and frequency. The staff finds that Generic Safety Issue 83 is adequately addressed.

In RAI 6.4-5, RAI 6.4-6, and RAI 6.4-18, the RAIs requested editorial changes to the DCD to correct discrepancies. RAI 6.4-5, RAI 6.4-6, and RAI 6.4-18 were being tracked as open items in the SER with open items. The staff reviewed the responses to these RAIs, including the proposed DCD changes, and finds them acceptable. The staff confirmed that the applicant had incorporated these changes in DCD Tier 2, Revision 7. Therefore RAI 6.4-5, RAI 6.4-6, and RAI 6.4-18 are resolved.

6.4.4 Conclusions

The staff finds that the ESBWR control room habitability systems meet the requirements of SRP Section 6.4 and associated guidance and regulations. There is reasonable assurance that passive cooling features will be sufficient to limit the control room environment temperatures under the summer and winter design conditions to a range that is acceptable for equipment and operator performance.

6.5 Atmosphere Cleanup System

6.5.1 Regulatory Criteria

The atmosphere cleanup system is needed to mitigate the radiological consequences of postulated DBAs by removing fission products from the containment atmosphere that may be released from the reactor primary coolant system in the event of an accident and to meet the radiological consequence evaluation factors specified in 10 CFR 52.47(a)(2) and GDC 19.

The staff's bases its acceptance criteria for the atmosphere cleanup systems on the relevant requirements of the following regulations:

- GDC 19, as it relates to systems being designed for habitability of the control room during and following postulated DBAs

- GDC 41, as it relates to the design of systems to be used for containment atmosphere cleanup during and following postulated DBAs

- GDC 42 and GDC 43, as they relate to the inspection and testing of the systems

- GDC 61, "Fuel storage and handling and radioactive material," as it relates to the design of systems for radioactivity control

- 10 CFR 50.34(a)(1), as it relates to the radiological consequence evaluation factors specified for the exclusion area boundary and the low-population zone

The staff reviewed DCD Tier 2, Revision 9, Section 6.5, in accordance with the following SRP sections:

- Section 6.5.1
- Section 6.5.3
- Section 6.5.5

NUREG–0800, Revision 2, dated July 1981, Section 6.5.2 and Section 6.5.4 are not used because the ESBWR design does not include either a safety-related containment spray system or an ice condenser.

6.5.2 Summary of Technical Information

Containment

The ESBWR design does not provide an active containment atmosphere cleanup system. Instead, the design relies on natural aerosol removal processes, such as gravitational settling and plateout on containment internal structure surfaces through diffusiophoresis and thermophoresis. The containment structure is a reinforced concrete cylindrical structure that encloses the RPV and its related systems and components and has an internal steel liner providing the leaktight containment boundary. The ESBWR containment is designed to a maximum allowable design leak rate of 0.35 wt% per day. The applicant stated that 0.01 wt% per day of a 0.35 wt% overall containment leak is assumed to leak through the PCCS into the air space directly above the PCCS and subsequently leak directly to the environment without mixing with the RB atmosphere.

Passive Containment Cooling System

The PCCS is designed to remove decay heat and fission products from the containment atmosphere following a postulated DBA. The PCC heat exchangers receive a steam-gas mixture and airborne fission products from the drywell atmosphere, condense the steam, and return the condensate, with condensed fission products, to the RPV though the GDCS pools. The noncondensables, including noble gases and volatile fission products, are drawn to the suppression pool through a submerged vent line driven by the differential pressure between the drywell and wetwell. The noncondensables will become airborne into the wetwell air space and flow back into the drywell during vacuum breaker openings.

Reactor Building

The RB is a reinforced concrete structure, which forms an envelope completely surrounding the containment and is designed to seismic Category 1 criteria. The RB does not have an atmospheric cleanup system. The RBVS isolation dampers will be tested as described in DCD Tier 1, Section 2.16.2, to support the radiological consequence analysis performed in Chapter 15 of this report. During normal plant operation, the potentially contaminated areas of the RB are maintained at a slightly negative pressure, relative to adjoining areas, by a nonsafety-related RB HVAC system. Following a postulated DBA, the RB HVAC system is automatically isolated. The RB has a design maximum leakage of 141.6 l/s (300 cfm). The applicant stated that the RB envelope is not intended to provide a leaktight barrier against a radiological fission product release. The applicant further stated that the RB will be periodically tested to ensure that the leakage rates assumed in the radiological consequence analyses are met.

Suppression Pool

The ESBWR design provides, among other things, a suppression pool to condense steam and remove fission products following a postulated DBA. The applicant did not take credit for suppression pool scrubbing in the bounding accident scenario considered, as it is a low pressure event. The flow through the SRVs is negligible for low pressure events.

Control Room Emergency Filter Unit

In DCD Tier 2, Revision 3, the applicant described the control room EFU (CREFU), which is an ESF atmosphere cleanup system to prevent the intrusion of fission products into the main CRHA and to pressurize the control room with nonradioactive outside air following postulated DBAs. The CREFU, a subsystem of the CB HVAC system, is a safety-related system and is located in the CB. The CB is designed to seismic Category 1 criteria. The CREFU replaces the passive control room emergency air breathing system provided in previous revisions to the DCD.

The CREFU consists of two redundant trains, each with a pre-filter, HEPA filter, .1 m (4-in.) deep charcoal adsorber, and post-filter to remove fission products and to pressurize the control room to prevent any inleakage of radioactive material into the control room following postulated DBAs. Two redundant trains, which are physically and electrically redundant and separated, provide single active failure protection for the CREFU. The CREFU equipment and components are designed to seismic Category 1 and are located in a seismic Category 1 structure. The CREFU trains are operable during loss of preferred power, loss of onsite ac power, or SBO, and they are designed, constructed, and tested to meet the requirements of RG 1.52. The system will be automatically activated by high radioactivity in the MCR air supply duct or can be activated manually from the MCR.

Drywell Spray System

The ESBWR design includes a nonsafety-related drywell spray system for severe accident management to aid in postaccident recovery or to mitigate the effects of a severe accident. The nonsafety-related drywell spray system is not credited for removal of fission products in the radiological consequence evaluation.

6.5.3 Staff Evaluation

Section 6.2.1 of this report addresses the staff's evaluation of containment performance.

Section 6.2.3 of this report addresses the staff's evaluation of the applicant's assumptions related to RB leakage and mixing and the RB functional design.

Section 15.4.3 of this report presents the staff's evaluation of the removal of fission products by the PCCS as a means for meeting the radiological consequence evaluation factors in 10 CFR 50.34 (a)(1) and GDC 19. Section 6.2.2 of this report provides the staff's evaluation of the removal of decay heat by the PCCS.

In performing its independent confirmatory radiological consequence analysis, the staff used the MELCOR computer code, along with the ESBWR design specifics, to estimate fission product transport and removal by these passive systems. Section 15.4.3 of this report presents the staff's evaluation on the removal of fission products by these passive systems and structures, as a means for meeting the radiological consequence evaluation factors in 10 CFR 50.34(a)(1) and GDC 19.

Sections 6.4 and 9.4.1 of this report provide the staff's evaluation of whether the CREFU meets the requirements of GDC 19. Section 15.4.4.3.2.4 of this report summarizes the radiological consequence analysis, using the CREFU for the control room habitability following postulated DBAs as a means for meeting the radiological consequence evaluation factors in GDC 19.

6.5.4 Conclusions

Based on the staff's review of the information provided by GEH, the staff concludes that the passive atmosphere cleanup systems provided in the ESBWR design, which are intended to mitigate the radiological consequences of postulated DBAs by removing fission products from the containment atmosphere that may be released from the reactor primary coolant system in the event of an accident, meet the radiological consequence evaluation factors specified in 10 CFR 52.47(a)(2) and GDC 19.

6.6 Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

6.6.1 Regulatory Criteria

The staff reviewed ESBWR DCD Tier 2, Revision 9, Section 6.6, in accordance with SRP Section 6.6, Revision 2, issued March 2007. This SRP section states that the requirements for periodic inspection and testing of Class 2 and 3 systems in GDC 36, 37, 39, 40, 42, 43, 45, and 46 are specified in 10 CFR 50.55a and detailed in Section XI of the ASME Code as described below.

- 10 CFR 50.55a contains preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.

- GDC 36 requires that the design of the ECCS permit appropriate periodic inspection of important safety components, such as spray rings, in the RPV.

- GDC 37 requires that the design of the ECCS permit appropriate testing to ensure structural integrity, leaktightness, and the operability of the system.

- GDC 39 requires that the design of the containment heat removal system permit inspection of important components, such as the torus and spray nozzles, to ensure the integrity and capability of the system.

- GDC 40 requires that the design of the containment heat removal system permit appropriate periodic pressure and functional testing to ensure the structural and leaktight integrity of its components, the operability and performance of the active components of the system, and the operability of the system as a whole.

- GDC 42 requires that the design of the containment atmospheric cleanup system permit appropriate periodic inspection of components such as filter frames and ducts to ensure integrity and capability of the system.

- GDC 43 requires that the design of the containment atmospheric cleanup system permit appropriate periodic pressure and functional testing to ensure the structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.

- GDC 45, “Inspection of cooling water system,” requires that the design of the cooling water system permit appropriate periodic inspection of important components, such as heat exchangers, to ensure the integrity and capability of the system.

- GDC 46, “Testing of cooling water system,” requires that the design of the cooling water system permit appropriate pressure and functional testing to ensure the structural and leaktight integrity of its components, the operability and performance of the active components of the system, and the operability of the system as a whole.

ASME Class 2 and 3 components rely upon these design provisions to allow performance of an ISI. Compliance with these GDC ensures that the design of the safety systems will allow access to important components, so that periodic inspections can detect degradation, leakage, signs of mechanical or structural distress caused by aging, and fatigue or corrosion, before the ability of these systems to perform their intended safety functions is jeopardized.

6.6.2 Summary of Technical Information

DCD Tier 2, Revision 9, Section 6.6, states that the ESBWR meets the requirements for periodic inspection and testing of Class 2 and 3 systems in GDC 36, 37, 39, 40, 42, 43, 45, and 46, as specified, in part, in 10 CFR 50.55a and as detailed in Section XI of the ASME Code. The ESBWR meets the acceptance criteria in SRP Section 6.6, Revision 1, by conforming to the ISI requirements of the aforementioned GDC and 10 CFR 50.55a for the areas of review described in Section I of the SRP.

The applicant stated that all items within the Class 2 and 3 boundaries provide access for the examinations required by ASME Code, Section XI, Subarticles IWC-2500 and IWD-2500.

The physical arrangement of piping, pumps, and valves provides personnel access to each weld location for the performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for the performance of visual (i.e., VT-3) examinations. Working platforms in some areas facilitate the servicing of pumps and valves. Removable thermal insulation is provided on welds and components that require frequent access for examination or are located in high-radiation areas. The design of weld locations permits ultrasonic examination from at least one side and access from both sides, where component geometry permits.

The personnel performing examinations shall be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with an industry-accepted program for implementation of ASME Code, Section XI, Appendix VIII. Circumferential welds in high-energy piping between the CIVs shall be 100-percent volumetrically examined at each inspection interval.

Piping systems that are ASME Code, Section III, Code Class 1, 2, and 3, as well as nonsafety-related piping, and components described in NRC GL 89-08 that are determined to be susceptible to erosion or corrosion shall be subject to a program of nondestructive examination (NDE) to verify a system's structural integrity. The examination schedule and methods shall be determined in accordance with the Electric Power Research Institute (EPRI) guidelines in Nuclear Safety Analysis Center (NSAC)-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," issued April 1999, which satisfy NRC GL 89-08, or the latest revision approved by the NRC (or an equally effective program) and the applicable rules of ASME Code, Section XI.

The COL licensee will be responsible for developing the site-specific preservice inspection (PSI) and ISI program plans, which will be based on the ASME Code, Section XI, edition and addenda approved in 10 CFR 50.55a(b), 12 months before initial fuel load. The COL applicant is responsible for providing a full description of the PSI/ISI programs and augmented inspection programs for Class 2 and 3 components and piping by supplementing, as necessary, the information in DCD Tier 2, Revision 9, Section 6.6. The COL applicant will provide milestones for program implementation (COL Information Item 6.6-1-A). The COL applicant is also responsible for providing a full description of PSI/ISI, and design activities for components that are not included in the referenced design, to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and dissimilar metal welds during ISI (COL Information Item 6.6-2-A).

6.6.3 Staff Evaluation

The staff's evaluation of the ISI program description of ASME Code Class 2 and 3 components is contained in the following six sections—(1) components subject to inspection, (2) accessibility, (3) examination categories and methods, (4) evaluation of examination results, (5) system pressure tests, and (6) augmented ISI to protect against postulated piping failure.

6.6.3.1 *Components Subject to Inspection*

The definitions of ASME Code Class 2 and 3 components and systems subject to an ISI program are acceptable if they agree with the NRC quality group classification system (RG 1.26) or the definitions in Article NCA-2000 of Section III of the ASME Code. Section 3.2.2 of this report contains the staff's evaluation of the applicant's classification of components.

6.6.3.2 *Accessibility*

The applicant indicated that, in the ESBWR design, all items within the Class 2 and 3 boundaries provide access for the examinations required by ASME Code, Section XI, Subarticle IWC-2500 and IWD-2500.

The staff issued RAI 6.6-1, RAI 6.6-2, RAI 6.6-3, RAI 6.6-4, RAI 5.2-51, RAI 5.2-53, RAI 5.2-54, RAI 5.2-57, and RAI 5.2-58 regarding the accessibility of components to inspections required by ASME Code, Section XI, and 10 CFR 50.55a. The staff developed RAI 5.2-62, which supersedes the aforementioned RAIs, regarding the accessibility and inspectability of welds and components. In RAI 5.2-62, the staff requested that the applicant modify the DCD to (1) specify the inspection methods that are practical to use for an ISI of welds in ASME Code Class 1 and 2 austenitic and dissimilar metal welds, and (2) add COL information items to Sections 5.2.4 and 6.6 to ensure that a COL applicant referencing the DCD will provide a detailed description of its plans to incorporate, during design and construction, access to piping systems to enable NDE of such welds during an ISI.

By way of background, the staff understands that materials selected for use in the ESBWR ASME Code Class 1 and 2 austenitic and dissimilar metal welds are not expected to encounter SCC or an appreciable amount of other forms of degradation, based on currently available information. However, the staff notes that SCC was not expected in previously built pressurized-water reactors and BWRs, based on information that was available at the time of their licensing and construction. Accordingly, the staff considers that the design of components should include provisions to enable NDE to detect future component degradation, such as SCC. This is a critical attribute of any new reactor design.

ASME Code, Section XI, as incorporated into 10 CFR 50.55a(g), currently allows for either ultrasonic or radiographic examinations of welds in ASME Code Class 1 and 2 piping systems. The staff requested that the applicant modify the DCD in Tier 1 to state that one or both of these types of examination are practical for ISI of austenitic and dissimilar metal welds. The staff notes that ultrasonic examination has advantages with respect to ALARA considerations and, with this change to the DCD, any design certification rule that might be issued for the ESBWR will preclude the granting of relief under 10 CFR 50.55a(g)(6) for ISI of such welds. The staff requested that the applicant confirm that austenitic or dissimilar metal welds in ASME Code Class 1 and 2 piping systems will be accessible for examination by either ultrasonic or radiographic examination, in accordance with the requirements of 10 CFR 50.55a(g)(3).

In support of these DCD changes, a COL applicant referencing the ESBWR design certification application should tell the staff how it plans to meet all access requirements during construction and operation, as required by 10 CFR 50.55a(g)(3)(i) and (ii). The staff notes that the PSI requirements are known at the time a component is ordered, and 10 CFR 50.55a(g) does not contain provisions for consideration of relief requests for impractical examinations during the construction phases of the component. The staff asked that the COL information items requested above reflect these considerations. The staff identified this issue in RAI 5.2-62. RAI 5.2-62 was being tracked as an open item in the SER with open items.

The applicant responded by letter and indicated that it would modify DCD Tier 2, Sections 5.2.4 and 6.6, to include a description of its design process to ensure that the accessibility of austenitic and dissimilar metal welds to perform UT or radiographic testing (RT). The staff reviewed the applicant's RAI response and modifications in DCD Tier 2, Revision 5, Sections 5.2.4 and 6.6, and found them to be unacceptable because they did not address the design's accessibility, taking into account operational and radiological concerns. The staff issued RAI 5.2-62 S01 and requested that the applicant address this issue.

In response, the applicant stated that it would modify DCD Tier 2, Sections 5.2.4 and 6.6, to address the staff's concerns. Section 5.2.4 of this report addresses the accessibility of Class 1 components. The applicant proposed the modifications below to DCD Tier 2, Section 6.6.2, which includes Tier 2* information, in lieu of the Tier 1 changes requested by the staff. Given that the COL applicant cannot depart from Tier 2* information without NRC approval, the staff considers that making the modifications in Tier 2* is acceptable.

> [The ESBWR design includes specific access requirements, in accordance with 10 CFR 50.55a(g)(3), to support preferred UT or optional RT examinations. The design of each component and system takes into account the NDE method, UT or RT, that will be used to fulfill preservice inspection and in-service inspection examination and will take into full consideration the operational and radiological concerns associated with the method selected to ensure that the performance of the required examination will be practical during commercial operation of the plant. Additionally, the design procedural requirements for the 3D layout of the plant include acceptance criteria regarding access for inspection equipment and personnel]*. However, with respect to any design activities for components that are not included in the referenced ESBWR certified design, it is the responsibility of the COL Applicant to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and DM welds during in-service inspection (COL Information Item 6.6-2-A)

The staff finds that the proposed modifications to DCD Tier 2, Section 6.6.2, discussed above, ensure that austenitic and dissimilar metal welds will be accessible to perform ASME Code-required inspections, taking into account operational and radiological concerns that could affect the practicality of the inspection method chosen for ISIs and PSIs. The staff subsequently confirmed that the applicant had made the above modifications to DCD Tier 2, Section 6.6. Based on the applicant's response, RAI 5.2-62 is resolved.

6.6.3.3 *Examination Categories and Methods*

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. Thus, the examination categories and methods specified in the DCD are acceptable if they agree with the requirements in Articles IWA-2000, IWC-2000, and IWD-2000 of Section XI of the ASME Code. The staff will review the COL applicant's description of its ISI program during the COL application review.

DCD Tier 2, Revision 9, Section 6.6.3.1, indicates that all of the items selected for inservice examination will receive a preservice examination, in accordance with ASME Code, Section XI, Subarticles IWC-2200 and IWD-2200, with the exception of the preservice examinations specifically excluded by the ASME Code. For the aforementioned exception to preservice examination, the applicant provides examples, such as the visual VT-2 examinations for Categories C-H and D-A.

DCD Tier 2, Revision 9, Section 5.2.4, indicates that the design regarding PSI is based on the requirements of ASME Code, Section XI, as specified in DCD Tier 2, Revision 9, Table 1.9-22. Table 1.9-22 indicates that the above-referenced code is the ASME Code, Section XI, 2001 Edition through the 2003 Addenda.

It appeared that the applicant has made references to the 1989 edition of ASME Code, Section XI, regarding examination Category D-A. The staff noted that, in other instances, the applicant also referenced examination categories from the 1989 ASME Code. In RAI 5.2-56, the staff requested that the applicant update references to examination categories that were apparently referenced from the 1989 ASME Code. Subsequently, in a response to RAI 5.2-56 the applicant responded that it corrected the applicable references. Given that GEH has indicated that the information it supplied is based on an updated ASME Code, Section XI, the staff requested, in RAI 6.6-8, that GEH modify DCD Tier 2, Section 6.6, to reference the appropriate examination categories for the 2001 Edition through the 2003 Addenda. The staff also requested that the applicant verify that it has reviewed DCD Tier 2, Sections 5.2.4 and 6.6 to ensure that all references to ASME Code, Section XI, are consistent with the 2001 Edition through the 2003 Addenda. The staff identified this issue in RAI 6.6-8. RAI 6.6-8 was being tracked as an open item in the SER with open items.

The applicant responded and stated that it would modify the incorrect examination categories listed in DCD Tier 2, Sections 6.6 and 5.2.4. The applicant also stated that a complete review of DCD Tier 2, Sections 5.2.4 and 6.6, was performed and appropriate changes would be made to DCD Tier 2, Section 5.2.4 and 6.6. The staff confirmed that the appropriate modifications, as discussed in the applicant's December 21, 2007, letter were made in DCD Tier 2, Revision 5. Based on the applicant's response, RAIs 6.6-8 and 5.2-56 are resolved.

The DCD Tier 2, Section 6.6.3.2.6, indicates that personnel performing ultrasonic examinations will be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for the implementation of ASME Code, Section XI, Division 1, Appendix VIII. The staff finds this acceptable, given that any industry-accepted program is required to meet Appendix VIII requirements, in accordance with the implementation requirements of 10 CFR 50.55a.

The staff requested information regarding the ISI requirements for ICs and PCCS heat exchangers (condensers), because it is not clear whether ASME Code requirements are sufficient to ensure that these components will be inspected in a manner that will provide reasonable assurance that degradation that may occur will be detected in a timely fashion and thus prevent component failure. The IC heat exchangers are ASME Code Class 2 and the PCCS heat exchangers are ASME Code Class MC.

In RAI 5.4-56, the staff requested that the applicant confirm that the method or technique for inspecting IC tubes is capable of detecting general wall thinning, pit-like defects, and SCC along the entire length of the tube. In RAI 5.4-58, the staff requested that the applicant discuss the results of inspections performed on Alloy 600 components in operating BWRs. RAI 5.4-56 and RAI 5.4-58 were being tracked as open items in the SER with open items. In response to RAI 5.4-58, the applicant indicated that modified Alloy 600 has been in service for a number of years but that it has not currently been inspected as part of a formal ISI program. In response to RAI 5.4-56, the applicant indicated that, because of the size of the IC tubes (nominal pipe size 2), they are exempt from volumetric and surface inservice examinations by ASME Code, Section XI, Paragraph IWC-1220, which exempts nominal pipe sizes 4 and smaller. The applicant indicated that the ICs are subject to leakage examination (i.e., VT-2) under ASME Code, Section XI. However, visual examination will only indicate whether the degradation has penetrated through wall (which would normally be detected through radiation monitoring techniques). There is a lack of long-term service experience (with inspection results), and the limitations of accelerated corrosion testing prevent fully simulating the range of variables that may exist in the field (and that may be pertinent to corrosion). Therefore, in RAI 5.4-58 S01, the staff requested additional information concerning the inspection and acceptance criteria for the IC tubes or justification for the lack of inspection requirements. RAI 5.4-58 S01 also requested that the applicant provide a response that addresses the original RAI 5.4-56, since visual inspections will not indicate whether the IC tubes have degraded by corrosion or mechanical mechanisms unless the degradation has penetrated through wall (at this point, the IC tubes may no longer have adequate integrity). In summary, the staff requested that the applicant provide the inspection and acceptance criteria for the IC tubes and confirm that volumetric inservice examination techniques exist for finding the forms of degradation that may affect the IC tubes. The staff identified these issues in RAI 5.4-56 and RAI 5.4-58. In response, the applicant stated that SCC is not plausible because of the IC pool temperature, control of water chemistry, use of modified Alloy 600, and lack of crevices in the IC heat exchanger assembly. The applicant also stated that the IC design takes general corrosion into account.

The staff identified two degradation mechanisms that could be of concern in the IC. They are SCC and general corrosion. With regard to SCC, the use of modified Alloy 600 greatly reduces the risk. Pressure boundary welds, such as IC header and tube-to-header welds, are full penetration welds that do not contain crevices that could be initiation sites for SCC to occur. The low normal operating temperature of the IC pool and water chemistry controls, coupled with the use of niobium-modified Alloy 600, make the possibility of SCC unlikely. In the event that leakage were to occur because of a through wall flaw, it would be detected by radiation leakage monitoring equipment. General corrosion of modified Alloy 600 is considered negligible in the IC environment. In addition, the applicant stated that the IC design takes into account general corrosion and its effects. Based on the resistance of modified Alloy 600 to SCC in the BWR environment, the lack of crevices to act as SCC initiation sites and the low operating temperature of the IC pool, the staff does not consider augmented inservice examinations necessary, beyond current ASME Code requirements. Based on the applicant’s response, RAI 5.4-56 and RAI 5.4-58 are resolved.

Since the limitations of accelerated corrosion testing also apply to the PCCS heat exchanger tubes, the staff requested similar information for the PCCS heat exchanger. In addition, the staff requested clarification to determine whether the cracking that occurred in earlier ICs could occur in the PCCS heat exchanger. The staff identified this issue in RAI 5.4-57. RAI 5.4-57 was being tracked as an open item in the SER with open items. In response, the applicant stated that the PCCS heat exchangers are fabricated from 304L stainless steel, immersed in deionized water at ambient pressure and temperature, and only used post-LOCA. In addition, the post-LOCA environment is flowing steam at a maximum of 171 degrees C (340 degrees F) for 72 hours. The applicant stated that, under these conditions, corrosion of stainless steel is extremely limited, as are other forms of material degradation. Cracking of stainless steel in the PCCS heat exchangers is not expected to occur as it has in the past in stainless steel IC tubes, because the PCCS uses low-carbon stainless steels and is submerged in chemically controlled deionized water at ambient temperature and pressure for essentially its entire life. In addition, the applicant indicated that, because of the operating conditions and environment, no augmented inspections are necessary. The staff agrees that general corrosion of the PCCS heat exchangers fabricated from low-carbon stainless steel in the expected environment will be negligible. The potential for SCC is all but eliminated through the use of low-carbon stainless steel, the absence of crevices in the tube-to-header full penetration welds, extremely low normal operation temperature and pressure, a chemically controlled deionized water environment, and limited use for these components under post-LOCA conditions. Based on the corrosion resistance of 304L in the PCCS pool environment, which includes chemically controlled deionizer water, low operating temperature and low operating pressure, the staff finds that 304L will not be susceptible to stress-corrosion cracking, and therefore, the staff does not consider augmented inservice examinations necessary, beyond current ASME Code requirements. Based on the applicant's response, RAI 5.4-57 is resolved.

In DCD Revision 7, the applicant modified DCD Tier 2, Table 6.1-1 to change the material used for the PCCS heat exchanger tubes from 304L to XM-19. XM-19, also known as NITRONIC 50, is a nitrogen strengthened austenitic stainless steel which has a higher yield and tensile strength than 304L stainless steel. In addition, XM-19 has superior corrosion resistance to 304L in the PCCS operating environment and is acceptable for use in accordance with ASME Section III materials specifications requirements. Therefore, degradation of XM-19 used in the PCCS heat exchanger tubes is expected to be negligible for the design life of the plant. The staff therefore finds the applicant's use of XM-19 PCCS heat exchanger tubes acceptable, and no augmented inservice inspections are required.

6.6.3.4 *Examination Intervals*

The required examinations and pressure tests must be completed during each 10-year interval of service, hereafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of ASME Code, Section XI, Article IWA-2000, concerning inspection intervals.

DCD Tier 2, Revision 9, Section 6.6.4, discusses inspection intervals for ASME Code Class 2 and 3 systems. Subarticles IWA-2400, IWC-2400, and IWD-2400 of ASME Code, Section XI, define inspection intervals. The inspection intervals specified for the ESBWR components are consistent with the definitions in Section XI of the ASME Code and, therefore, are acceptable.

6.6.3.5 *Evaluation of Examination Results*

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. GEH indicated that examination results are evaluated in accordance with ASME Code, Section XI, Article IWC-3000, for Class 2 components, with repairs based on the requirements of Article IWA-4000. Examination results are evaluated in accordance with ASME Code, Section XI, Article IWD-3000, for Class 3 components, with repairs based on the requirements of Article IWA-4000. The GEH description of the evaluation of examination results is consistent with ASME Code, Section XI, and meets the acceptance criteria in SRP Section 6.6, Section II.5, and is therefore acceptable.

6.6.3.6 *System Pressure Tests*

DCD Tier 2, Revision 9, Sections 5.2.4.6 and 6.6.6, reference certain portions of ASME Code, Section XI, Articles IWA-5000, IWB-5000, IWC-5000, and IWD-5000, in the description of system leakage and hydrostatic pressure tests for ASME Code Class 1, 2, and 3 systems. In RAI 5.2-65, the staff requested that the applicant modify DCD Tier 2, Sections 5.2.4.6 and 6.6 to clarify that system leakage and hydrostatic pressure tests will meet all requirements of ASME Code, Section XI, Articles IWA-5000, IWB-5000, IWC-5000, and IWD-5000. RAI 5.2-65 was being tracked as an open item in the SER with open items. The applicant responded that it would modify DCD Tier 2, Section 6.6.6, to state that the requirements of IWA-5000 and IWC-5000 will be met for Class 2 components, and the requirements of IWA-5000 and IWD-5000 will be met for Class 3 components. The applicant's response addressed requirements for Class 1 components, and Section 5.2.4 of this report discusses them. The staff reviewed Revision 5 to the DCD and verified that the appropriate modifications were made to Section 6.6.6. Based on the applicant's response, RAI 5.2-65 is resolved.

6.6.3.7 *Augmented Inservice Inspection To Protect against Postulated Piping Failure*

The augmented ISI program for high-energy fluid systems piping between CIVs is acceptable, if ISI examinations completed during each inspection interval provide a 100-percent volumetric examination of circumferential and longitudinal pipe welds with the boundary of these portions of piping. DCD Tier 2, Revision 9, Section 6.6.7, indicates that high-energy piping (as defined in DCD Tier 2, Section 3.6.2) between CIVs is subject to additional inspection requirements. Circumferential welds shall be 100-percent volumetrically examined at each inspection interval. The piping in these areas is seamless, thereby eliminating longitudinal welds. The applicant's augmented ISI program to protect against postulated pipe failure is consistent with SRP Section 6.6 and is, therefore, acceptable.

BL 80-08 "Examination of Containment Liner Penetration Welds," identifies NRC concerns related to UT of primary piping containment penetration fluid-head (integral fitting) to outer sleeve welds, using backing bars, which form part of the containment pressure boundary. In RAI 20.0-5, the staff requested that the applicant address the NRC concerns identified in BL 80-08. In response, the applicant stated that backing bars are not used in flued-head containment penetration assemblies or other penetration sleeves and process piping. The applicant also stated that it would modify DCD Tier 2, Table 3.8-5, accordingly. The staff reviewed DCD Tier 2, Revision 5, and confirmed that the applicant had made the appropriate modifications. Based on the above, the staff has finds that the ESBWR design appropriately addresses the NRC concerns identified in BL 80-08 and RAI 20.0-5 is resolved.

6.6.3.8 *Augmented Erosion/Corrosion Inspection Program*

BL 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," dated July 9, 1987, requested that operating reactor licensees submit information concerning their programs for monitoring the thickness of pipe walls in high-energy, single-phase and two-phase carbon steel piping systems. The staff subsequently issued GL 89-08, requiring operating reactor licensees to verify implementation of formalized procedures or administrative controls to ensure continued long-term implementation of the erosion and corrosion monitoring program for piping and components. The ESBWR design requires COL applicants to develop appropriate long-term monitoring for potential wall thinning of high-energy piping by erosion and corrosion, as described in GL 89-08. In addition, COL Information Item 6.6-1-A requires COL applicants to provide a full description of augmented inspection programs and milestones for program implementation. The staff therefore finds that BL 87-01 and GL 89-08 are resolved for the ESBWR design. GL 89-08 is discussed further below.

As described in GL 89-08, an appropriate long-term monitoring program for potential wall thinning of high-energy piping by erosion and corrosion must be implemented. The applicant has indicated that all piping systems that are ASME Code, Section III, Code Class 1, 2, and 3, as well as nonsafety-related piping, and components described in GL 89-08 that are determined to be susceptible to erosion or corrosion shall be subject to NDE to verify system integrity. The applicant further stated that the examination schedule and methods shall be determined in accordance with EPRI guidelines in NSAC-202L-R2 or the latest revision approved by the NRC (or an equally effective program). The staff finds this acceptable, because it meets current NRC guidance. To verify that COL applicants will develop an appropriate long-term monitoring program for potential wall-thinning of high-energy piping by erosion or corrosion before plant startup, the staff requested, in RAI 5.2-64, that the applicant revise DCD Tier 2, Sections 5.2.4 and 6.6 to include a COL applicant action item to provide a detailed description of the PSI/ISI and augmented inspection programs and to provide milestones for their implementation. The applicant appropriately addressed this issue and the staff's detailed analysis of this response is found in Section 6.6.3.9 of this report.

6.6.3.9 *Combined License Information*

DCD Tier 2, Section 6.6.11, states that "The unit specific PSI/ISI Plan includes detailed plant information and is the responsibility of the COL holder as per Subsection 6.6.10." In RAI 5.2-64, the staff requested that the applicant revise DCD Tier 2, Sections 5.2.4 and 6.6 to include a COL information item to describe the PSI/ISI and augmented inspection programs and to provide milestones for their implementation. The staff was concerned that the GEH reference to the COL holder (the Licensee) does not make it clear that the COL applicant must provide a description of its PSI/ISI and augmented inspection programs with commitments for scheduled implementation of those programs identified in the COL application. It is understood that the COL licensee will fully develop and implement the actual programs. However, the COL applicant must fully describe the PSI/ISI and augmented inspection programs to allow the staff to make a reasonable assurance finding of acceptability. The staff was tracking RAI 5.2-64 as an open item in the SER with open items.

The applicant responded to RAI 5.2-64 and indicated that it would modify DCD Tier 2, Section 6.6.11, to address the staff's concerns. The staff reviewed DCD Tier 2, Revision 5, and confirmed that the appropriate modifications were made to Section 6.6.11. COL Information Item 6.6-1-A now states that the COL applicant is responsible for providing a full description of the PSI/ISI and augmented inspection programs for Class 2 and 3 components and piping, by supplementing, as necessary, the information in Section 6.6. The COL applicant will also provide milestones for program implementation (Section 6.6). Based on the applicant's response, RAI 5.2-64 is resolved.

COL Information Item 6.6-2-A states that the COL applicant is responsible for developing a plan and providing a full description of its use during construction, PSI, ISI, and for design activities for components that are not included in the referenced certified design, to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and dissimilar metal welds during ISIs (Section 6.6).

6.6.4 Conclusions

The staff concludes that the ESBWR program for Code Class 2 and 3 components is acceptable and meets the inspection and pressure-testing requirements of 10 CFR 50.55a, as detailed in ASME Code, Section IX, and therefore satisfies the applicable requirements of GDC 36, 37, 39, 40, 42, 43, 45, and 46.

7.0 INSTRUMENTATION AND CONTROLS

7.0 Instrumentation and Controls – Introduction

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff review of the instrumentation and control (I&C) portion of the GE-Hitachi Nuclear Energy (GEH) application for the economic simplified boiling-water reactor (ESBWR). The staff review of the ESBWR I&C is part of the overall design certification review conducted by the staff under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." This review is conducted in accordance with NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," March 2007 (hereafter referred to as the SRP) Chapter 7, Revision 5. Consistent with SRP Chapter 7, the review used 10 CFR 50.55a(h), which requires that applications for design certification filed on or after May 13, 1999, meet the requirements for safety systems in Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations", and the correction sheet dated January 30, 1995. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," provides guidance on applying the safety system criteria to computer-based systems. The NRC endorsed IEEE Std 7-4.3.2-2003 in Regulatory Guide (RG) 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."

7.0.1 Method of Review

Consistent with SECY-92-053, "Use of Design Acceptance Criteria (DAC) During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992, Tier 2 of the ESBWR design control document (DCD) provides limited design details for I&C systems. Accordingly, DCD Tier 1 provides associated design acceptance criteria (DAC) that a combined license (COL) applicant or licensee would follow to complete the design detail. (DAC are a type of inspections, tests, analyses, and acceptance criteria [ITAAC] marked with {{Design Acceptance Criteria}} labels.)

The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, for a limited number of technical areas (e.g., digital I&C, human factors engineering [HFE], and piping), in making a final safety determination to support a design certification. The acceptance criteria for the DAC should be measurable, testable, or subject to analysis using pre-approved methods, and should be verified as part of the ITAAC used to demonstrate that the as-built facility conforms to the certified design. Thus, the acceptance criteria for DAC are specified, together with the related ITAAC, in DCD Tier 1, and both are part of the design certification. The DAC and the ITAAC, when met, ensure that the completed design and as-built plant conform to the design certification.

The safety basis of I&C systems in the ESBWR design can be divided into hardware and software aspects.

The hardware aspects are as follows:

- The applicant committed to comply with IEEE Std 603 and other applicable requirements.
- The applicant identified high level functional requirements.

- The applicant provided DAC to verify and confirm that the completed design meets IEEE Std 603 requirements.

The software aspects are as follows:

- The applicant identified and committed to a program to implement a software development process.

- The applicant provided DAC to verify and confirm the following:
  - Acceptable plans were prepared to control software development activities.
  - The plans were followed in an acceptable software life cycle.
  - The process produced acceptable design outputs.

The software and hardware design process is discussed below.

When a DCD includes DAC instead of providing design detail, the DCD should include the associated design processes that the COL applicant will use to complete the design. For software, the design includes a software design process described in NEDO-33226 (NEDE-33226P), “ESBWR – Software Management Program Manual” (SMPM), and NEDO-33245 (NEDE-33245P), “ESBWR – Software Quality Assurance Program Manual” (SQAPM). For hardware, the design process consists of commitments that the safety I&C systems are designed to follow IEEE Std 603 as documented in DCD Tier 1, Section 2.2.15, and that the safety I&C systems are qualified to meet the requirements documented in DCD Tier 2, Sections 3.10 and 3.11 and DCD Tier 1, Section 3.8.

Sections 7.1 through 7.8 of this report document the staff evaluation of the I&C design. The evaluation includes a verification of aspects of the design against the criteria in the respective SRP sections. The evaluation also includes verification that I&C system functions are compatible with the applicant’s accident analyses. This involved staff verification that the system events described in DCD Tier 2, Revision 9, Chapter 15, are consistent with the initiating actions described in DCD Tier 2, Revision 9, Chapter 7. In the necessary sections of this report, the evaluation will refer to IEEE Std 603, Section 5.5, as required by 10 CFR 50.55a(h)(3). This criterion requires that the safety systems accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. As explained below, the staff finds that I&C system functions are consistent with the events described in the accident analyses.

7.0.2 Documents for Instrumentation and Control Review

The staff’s evaluation includes Tier 1 and Tier 2 of the DCD. DCD Tier 2, Revision 9, Chapter 7, describes the primary I&C systems of the design.

- DCD Tier 2, Section 7.1, describes I&C system architecture and distributed control and information system (DCIS). Section 7.1 also discusses the conformance with regulatory requirements, industry codes, and standards.

- DCD Tier 2, Section 7.2, discusses the I&C aspects of the reactor trip function.

- DCD Tier 2, Section 7.3, addresses the I&C aspects of the engineered safety feature (ESF) actuation.

- DCD Tier 2, Section 7.4, discusses the systems in the design that are required for safe shutdown.

- DCD Tier 2, Section 7.5, discusses post accident monitoring (PAM), containment monitoring, and radiation monitoring systems.

- DCD Tier 2, Section 7.6, discusses interlocks important to safety.

- DCD Tier 2, Section 7.7, describes control systems in the design.

- DCD Tier 2, Section 7.8, addresses the Anticipated Transient without Scram (ATWS) mitigation function and defense against common cause failures (CCFs) within safety system designs.

- DCD Tier 2, Sections 7.1, 7.2, and 7.3, address data communications for the design.

In DCD Tier 1, Revision 9, the following sections address the I&C-related design commitments including the DAC/ITAAC:

- DCD Tier 1, Section 2.2, (includes Sections 2.2.1 through 2.2.16)
- DCD Tier 1, Section 3.2
- DCD Tier 1, Section 3.3
- DCD Tier 1, Section 3.7
- DCD Tier 1, Section 3.8

The applicant also submitted the following licensing topical reports (LTRs) to support the design certification review:

- NEDO-33251, “ESBWR I&C Defense-in-Depth and Diversity Report,” Revision 3

- NEDO-33304 (NEDE-33304P), “ESBWR Setpoint Methodology,” Revision 4

- NEDO-33295 (NEDE-33295P), “ESBWR Cyber Security Program Plan,” Revision 2

- NEDO-33226 (NEDE-33226P), "ESBWR – Software Management Program Manual,” Revision 5

- NEDO-33245 (NEDE-33245P), "ESBWR – Software Quality Assurance Program Manual," Revision 5

7.1 Introduction

This section documents the staff’s general evaluation of the DCIS (Section 7.1.1). It also documents the staff’s evaluation of nonsystem-based topics, including (1) software development activities (Section 7.1.2), (2) assessment of diversity and defense-in-depth (D3) (Section 7.1.3), (3) setpoint methodology (Section 7.1.4), (4) data communication systems (Section 7.1.5), and (5) secure development and operational environment (SDOE) (Section 7.1.6). Each section identifies the specific SRP section and the associated review criterion used in the review.

7.1.1 General Distributed Control and Information System Description

The I&C system uses the distributed digital system to perform plant-protection and safety monitoring functions, as well as control functions. The staff reviewed the DCIS in accordance with SRP Section 7.1. The staff used the acceptance criteria in SRP Table 7-1 and SRP Appendix 7.1-A to verify compliance with the applicable regulations, as directed by SRP Section 7.1. The staff also used SRP Appendices 7.1-C and 7.1-D to verify that DCD Tier 2, Revision 9, addressed all of the criteria listed in IEEE Std 603, as required by 10 CFR 50.55a(h)(3).

As identified in SRP Table 7-1 and in Sections 7.1 through 7.8 of this report, not all regulations and acceptance criteria listed below apply to each I&C system important to safety. However, Sections 7.1.1.1 and 7.1.1.3 of this report list and evaluate each regulation and acceptance criteria applicable to the DCIS. Sections 7.2 through 7.8 of this report focus on those acceptance criteria that the corresponding SRP section indicates should be emphasized. Sections 7.2 through 7.8 of this report also address system-specific criteria, as appropriate.

7.1.1.1 *Regulatory Criteria*

SRP Table 7-1, Section 1, identifies the following regulations as being applicable to I&C systems important to safety:

- 10 CFR 50.55a(a)(1), regarding quality standards for systems important to safety
- 10 CFR 50.55a(h)(3), regarding safety systems (i.e., IEEE Std 603)
- 10 CFR 50.34(f)(2)(v), regarding bypass and inoperable status indication
- 10 CFR 50.34(f)(2)(xi), regarding direct indication of relief and safety valve position
- 10 CFR 50.34(f)(2)(xiv), regarding containment isolation system
- 10 CFR 50.34(f)(2)(xvii), regarding accident monitoring instrumentation
- 10 CFR 50.34(f)(2)(xviii), regarding instrumentation for the detection of inadequate core cooling
- 10 CFR 50.34(f)(2)(xix), regarding instruments for monitoring plant conditions following core damage
- 10 CFR 50.34(f)(2)(xxiv), regarding central reactor vessel water level recording
- 10 CFR 50.62, regarding requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants
- 10 CFR 52.47(b)(1), regarding ITAAC for standard design certification

SRP Table 7-1, Section 2, identifies the following general design criteria (GDC), found in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," as being applicable to I&C systems important to safety:

- GDC 1, “Quality standards and records”
- GDC 2, “Design bases for protection against natural phenomena”
- GDC 4, “Environmental and dynamic effects design bases”
- GDC 10, “Reactor design”
- GDC 13, “Instrumentation and control”
- GDC 15, “Reactor coolant system design”
- GDC 16, “Containment design”
- GDC 19, “Control room”
- 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 25, “Protection system requirements for reactivity control malfunctions”
- GDC 28, “Reactivity limits”
- GDC 29, “Protection against anticipated operational occurrences”
- GDC 33, “Reactor coolant makeup”
- GDC 34, “Residual heat removal”
- GDC 35, “Emergency core cooling”
- GDC 38, “Containment heat removal”
- GDC 41, “Containment atmosphere cleanup”
- GDC 44, “Cooling water”

SRP Table 7-1, Section 3, identifies two items in the staff requirements memorandum (SRM) to SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Design,” dated April 2, 1993, as being applicable to I&C systems important to safety: (1) Item II.Q, “Defense Against Common-Mode Failure in Digital Instrumentation and Control Systems,” and (2) Item II.T, “Control Room Annunciator (Alarm) Reliability.”

SRP Table 7-1, Section 4, discusses RGs that provide acceptable methods for implementing the regulatory requirements for hardware and software features of digital systems important to safety. The RGs identified as being applicable to I&C systems important to safety are the following:

- RG 1.22, “Periodic Testing of Protection System Actuation Functions,” issued February 1972.

- RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems,” issued February 2010.

- RG 1.53, “Application of the Single-Failure Criterion to Safety Systems,” issued November 2003.

- RG 1.62, “Manual Initiation of Protective Actions,” issued June 2010.

- RG 1.75, Revision 3, “Criteria for Independence of Electrical Safety Systems,” issued February 2005.

- RG 1.97, Revision 4, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” issued June 2006.

- RG 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” issued December 1999.

- RG 1.118, Revision 3, “Periodic Testing of Electric Power and Protection Systems,” issued April 1995.

- RG 1.151, Revision 1, “Instrument Sensing Lines,” issued July 2010.

- RG 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," issued January 2006.

- RG 1.168, Revision 1, “Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” issued February 2004.

- RG 1.169, “Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” issued September 1997.

- RG 1.170, “Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” issued September 1997.

- RG 1.171, “Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” issued September 1997.

- RG 1.172, “Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” issued September 1997.

- RG 1.173, “Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants,” issued September 1997.

- RG 1.180, Revision 1, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,” issued October 2003.

- RG 1.189, Revision 2, “Fire Protection for Nuclear Power Plants,” issued October 2009.

- RG 1.204, “Guidelines for Lightning Protection of Nuclear Power Plants,” issued November 2005.

In addition to the RGs identified in SRP Table 7-1, Section 4, the following RG provides acceptable methods for implementing the regulatory requirements for hardware and software features of digital systems important to safety.

- RG 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” issued March 2007.

The following industry standards and documents are generally applicable to I&C systems that are important to safety but have not been incorporated into SRP Table 7-1:

- Electric Power Research Institute (EPRI) TR-107330, “Generic Requirements Specification for Qualifying a Commercially Available PLC [Programmable Logic Controller] for Safety-Related Applications in Nuclear Power Plants,” approved by the NRC on July 30, 1998.

- EPRI TR-102323-R1, “Guidelines for Electromagnetic Interference Testing in Power Plants,” approved by the NRC on April 17, 1996.

- EPRI TR-106439, “Guideline on Evaluation and Acceptance of Commercial - Grade Digital Equipment for Nuclear Safety Applications,” approved by the NRC in April 1997.

SRP Table 7-1, Section 5, identifies the applicability of SRP Branch Technical Positions (BTPs). DCD Tier 1, Revision 9, Table 7.1-1, noted that the design conforms to SRP Revision 4, BTP HICB-1 through HICB-21. The staff has compared the technical requirements between SRP Revision 4, BTP HICB-1 through HICB-21 and SRP Revision 5, BTP 7-1 through BTP 7-21. The staff finds that conformance with BTP HICB-1 through HICB-21 in SRP Revision 4 is equivalent to conformance with BTP 7-1 through BTP 7-21 in Revision 5 of SRP Table 7-1.

7.1.1.2 *Summary of Technical Information*

7.1.1.2.1 Instrumentation and Control Systems Overview

The I&C system for the ESBWR design is a DCIS. It is subdivided into the safety DCIS (Q-DCIS) and the nonsafety DCIS (N-DCIS). The Q-DCIS includes the reactor protection system (RPS), the neutron monitoring system (NMS), the independent control platform (ICP), and the safety system logic and control for the ESF actuation system (SSLC/ESF). The N-DCIS includes the diverse protection system (DPS), the balance of plant (BOP) systems, the plant investment protection (PIP) systems, the plant computer functions and workstations, and the severe accident mitigation system (deluge system). Tables 7-1 and 7-2 below summarize the safety category, the system architecture, and the subsystems in that family.

Table 7.1-1. Q-DCIS Overview.

System Families	Divisions	Subsystems	Platform

(RTIF-NMS)
Reactor Trip & Isolation Function - Neutron Monitoring System
	4 Independent Divisions	• RPS
		• (LD&IS) - Leak Detection and Isolation System
		  ○ Main Steam Isolation Valve (MSIV) Logic
		• NMS
		  ○ (SRNM) - Startup Range Neutron Monitoring System
		  ○ (PRNM) - Power Range Neutron Monitoring System
		• (CMS) - Containment Monitoring System
		  ○ (SPTM) - Suppression Pool Temperature Monitoring Function
			RTIF-NMS Platform

SSLC/ESF	4 Independent Divisions	<u>(ECCS)</u> - <u>Emergency Core Cooling System</u>
		• (ADS) - Automatic Depressurization System
		  ○ Safety Relief Valves (SRVs), Depressurization Valves (DPVs)
		• (GDCS) - Gravity-Driven Cooling System
		• (ICS) - Isolation Condenser System
		• (SLC) - Standby Liquid Control System
		<u>Non-ECCS</u>
		• LD&IS
		  ○ Non-MSIV Logic
		• (CRHS) - Control Room Habitability System
		• Safety-Related Information Systems
			SSLC/ESF Platform

Independent Logic Controllers	4 Independent Divisions	• ATWS Mitigation by SLC System • (VBIF) - Vacuum Breaker Isolation Function • High Pressure Control Rod Drive (HP CRD) Isolation Bypass Function (IBF) • ICS DPV Isolation Function (ICS DPVIF)	Independent Control Platforms
Safety-Related Information Systems	Per guidelines of RG 1.97	• PAM Instrumentation • (CMS) Containment Monitoring System • (PRMS) - Process Radiation Monitoring System	Included in the three platforms above

Table 7.1-2. N-DCIS Overview.

System Families	Redundancy	Subsystems	Platform
DPS	Triple Redundant	• Diverse RPS Logic • Diverse ECCS Logic • Diverse Containment Isolation • ATWS Mitigation Logic	Diverse DCIS Platform

Control Systems	Triple or Dual Redundant	• (NBS) - Nuclear Boiler System
		• (RC&IS) - Rod Control and Information System
		• (FWCS) - Feedwater Control System
		  ○ Feedwater Level Control System
		  ○ Feedwater Temperature Control System
		• (PAS) - Plant Automation System
		• (TGCS) - Turbine Generator Control System
		• (SB&PC) - Steam Bypass and Pressure Control System
		• NMS
		  ○ (AFIP) - Automatic Fixed In-Core Probe
		  ○ (MRBM) - Multichannel Rod Block Monitor
		• (CIS) - Containment Inerting System
			Diverse DCIS Platform

PIP DCIS Systems	Dual Redundant	• PIP-A & PIP-B	Diverse DCIS Platform
Plant Computer Functions (PCF)	Multiple Stations	• (SPDS) - Safety Parameter Display System • (MCRP) - Main Control Room Panel System • (AMS) - Alarm Management System • (OLPs) - Online Procedures • Technical Specification (TS) Monitoring • (3D-Monicore) - three-dimensional Monicore	Workstations
Nonsafety Information Systems	Per guidelines of RG 1.97	• PRMS • (ARMS) - Area Radiation Monitoring System • CMS	Workstations /Diverse DCIS Platform
Severe Accident Mitigation	Multiple PLCs	• Deluge System (a GDCS Subsystem)	PLC

7.1.1.2.2 Q-DCIS Overview

The Q-DCIS performs the safety control and monitoring functions. The Q-DCIS is organized into four physically and electrically isolated divisions. Each division is segmented into systems; segmentation allows, but does not require, the systems to operate independently of each other. The Q-DCIS major cabinets, systems, and functions are listed below.

The reactor trip and isolation function (RTIF) cabinets include the following:

- RPS
- MSIV functions of the LD&IS
- ATWS/SLC functions
- Vacuum breaker (VB) isolation function
- High pressure control rod drive (HP CRD) isolation bypass function
- ICS DPV isolation function
- Suppression pool temperature monitoring (SPTM) functions for RPS and CMS

Note: space considerations may require locating the ICP functions in separate cabinets.

The NMS includes the following:

- SRNM functions
- PRNM functions that include the following:
  - Local power range monitor (LPRM) functions
  - Average power range monitor (APRM) functions
  - Oscillation power range monitor (OPRM) functions

The SSLC/ESF system includes the following:

- ECCS functions that include the following:
  - ADS functions
  - GDCS functions
  - ICS functions
  - ECCS functions of the SLC system
- LD&IS functions (except the MSIV functions)
- CRHS functions
- Safety information systems

The Q-DCIS major components include the following:

- Fiber optic cable and hardwired network
- System control processors
- Non microprocessor-based logic
- Remote multiplexer units (RMUs)
- Load drivers (discrete outputs)
- Communication interface modules (CIMs)
- Video display units (VDUs)
- Main control room (MCR) wide-display and consoles that house the controls and monitoring

- Hard controls and indicators (for monitoring)
- Cabinets for housing devices such as power supplies

The RPS is the overall collection of instrument channels, trip logics, trip actuators, manual controls, and scram logic circuitry that initiate rapid insertion of control rods to shut down the reactor in situations that could result in unsafe operations. This action prevents or limits fuel damage, limits system pressure excursions, and thus minimizes the release of radioactive material. The RPS also establishes appropriate logic for different reactor operating modes, provides monitoring and control signals to other systems, and actuates alarms. The RPS hardware and logic are diverse from the SSLC/ESF logic, the ATWS mitigation logic, and the DPS logic. The RPS cabinet also houses the equipment that performs the SPTM functions for the CMS.

The NMS monitors neutron flux in the reactor core from the startup source range to beyond rated power. The NMS provides logic signals to the RPS to automatically shut down the reactor when a condition requires a reactor scram. The system provides an indication of neutron flux, which can be correlated with the thermal power level for the entire range of flux conditions that can exist in the core. The NMS comprises the following systems:

- The SRNM system monitors neutron flux levels from a very low-range power level to a power level above 15 percent of rated power. The SRNM system generates trip signals to prevent fuel damage resulting from abnormal positive reactivity insertion. The SRNM system generates both a high neutron flux trip and a high rate of neutron flux increase trip.

- The PRNM system includes the LPRM, APRM, and OPRM functions. The LPRM system provides the average power level of the reactor core and the OPRM system provides monitoring of neutron flux and core thermal-hydraulic instabilities. In the low end of the power range (1 percent to 15 percent), the SRNM and PRNM monitoring overlap.

- The AFIP is a nonsafety component of the NMS and does not provide information to the Q-DCIS. Its function is to calibrate the LPRMs by providing flux information to the 3D-Monicore system.

- The MRBM is a nonsafety component of the NMS and is completely isolated from the Q-DCIS by one-way optical fiber communications. Its function is to provide control rod blocks to the RC&IS to prevent violations of core thermal limits.

The SSLC/ESF system is the ESF actuation system for the design. The SSLC/ESF is the overall collection of instrument channels, trip logic, trip actuators, manual controls, and actuation logic circuitry that initiate protective actions to mitigate the consequences of design-basis events (DBEs). Input signals from redundant channels of safety instrumentation are used to make trip decisions to initiate the following accident mitigating functions:

- ECCS operation
- Leak detection, containment isolation, and radioactivity release barrier defense actuation
- MCR habitability functions

The ECCS provides emergency core cooling to respond to events that threaten the reactor coolant inventory. The ECCS comprises the ADS, the GDCS, the ICS, and the SLC system. The ADS resides within the NBS and comprises the SRVs, DPVs, and associated I&C. The ADS depressurizes the reactor to allow the low-head GDCS to provide makeup coolant to the reactor. The ADS logic resides in the SSLC/ESF portion of the Q-DCIS. The GDCS provides emergency core cooling once the reactor is depressurized. The GDCS is capable of injecting a large volume of water into the reactor pressure vessel (RPV) to keep the core covered for at least 72 hours following a loss-of-coolant accident (LOCA). The GDCS also performs a deluge function that drains the GDCS pools to the lower drywell in the event of a severe accident core-melt sequence. The GDCS deluge logic is separate and diverse from the Q-DCIS.

The ICS is designed to limit reactor pressure and prevent SRV operation following an isolation of the main steam lines. The ICS, together with the water stored in the RPV, provides sufficient reactor coolant volume to avoid automatic depressurization caused by a low reactor water level. The ICS is a safety system that removes reactor decay heat following reactor shutdown and isolation. The ICS logic resides on the SSLC/ESF portion of the Q-DCIS.

The SLC system performs dual functions. It provides additional coolant inventory to respond to a LOCA and serves as a backup method to bring the nuclear reactor to a subcritical condition and to maintain a subcritical condition as the reactor cools. The SLC logic resides on the SSLC/ESF and the ATWS/SLC portions of the Q-DCIS.

The LD&IS monitors leakage sources from the reactor coolant pressure boundary (RCPB) and automatically initiates closure of the appropriate valves that isolate the source of the leak. This action limits a coolant release from the RCPB and the release of radioactive materials to the environment. The LD&IS logic for the MSIVs resides on the RPS portions of the Q-DCIS and the non-MSIV isolation valve logic resides on the SSLC/ESF.

The CRHS provides a safe environment for the operators to control the nuclear reactor and its auxiliary systems during normal and abnormal conditions. The CRHS monitors the inlet ventilation air in the MCR habitability area and actuates logic to isolate and filter the control room habitability area (CRHA) upon detection of hazardous environmental conditions. The CRHS logic resides on the SSLC/ESF portion of the Q-DCIS.

The ATWS/SLC system provides a diverse means of reducing power excursions from certain transients and a diverse means of emergency shutdown. The ATWS mitigation logic, which uses the soluble boron injection capability of the SLC system as a diverse means of negative reactivity insertion, is implemented as safety logic. The ATWS/SLC logic also provides a feedwater runback (FWRB) signal to attenuate power excursions. The SLC may be initiated manually or automatically using the ATWS mitigation logic or the SSLC/ESF logic as an ECCS function. The SLC logic resides on the SSLC/ESF and ATWS/SLC RPS portions of the Q-DCIS. The nonsafety ATWS mitigation logic is implemented in the DPS.

The containment system wetwell-to-drywell vacuum breaker isolation function (VBIF) prevents the loss of long-term containment integrity upon detection of excessive vacuum breaker leakage. The VBIF is implemented by independent logic controllers.

The HP CRD isolation bypass function automatically bypasses the HP CRD injection isolation (intended to prevent the over-pressurization of the containment and loss of long-term containment integrity) to compensate for a failure of the GDCS to inject. The HP CRD isolation bypass function is implemented using the ICP, which is diverse from the RTIF-NMS and SSLC/ESF platforms and not susceptible to CCF.

The ICS DPV isolation function ensures that, upon detection of DPV open position, there is no loss of long-term containment integrity. This function is implemented in the ICP.

The passive containment cooling system (PCCS) functions to cool the containment following a rise in containment pressure and temperature without requiring any component actuation. The PCCS needs no electric power and does not have instrumentation, control logic, or power-actuated valves.

The SPTM system is part of the CMS. The system operates continuously during reactor operation. Should the suppression pool temperature exceed established limits, the system provides input both for a reactor scram and for automatic initiation of the suppression pool cooling mode of the fuel and auxiliary pool cooling system (FAPCS) operation.

Other CMS functions, some of which are nonsafety, include the monitoring of key containment fluid levels, radiation levels, pressures, concentrations, and dew point values. These parameters are monitored during both normal reactor operations and post accident conditions to evaluate the integrity and safe conditions of the containment. Abnormal measurements and indications initiate alarms in the MCR.

7.1.1.2.3 Nonsafety Distributed Control and Information System Overview

The N-DCIS components are redundant when they are needed to support power generation and are segmented into systems. The segmentation allows the systems to operate independently of each other. The N-DCIS cannot control any Q-DCIS component. The N-DCIS accepts one-way communication from the Q-DCIS so that the safety information can be monitored, archived, and alarmed seamlessly with the N-DCIS data.

The N-DCIS major systems and functions are described below.

The GENE systems include the following:

- Workstations
  - 3D Monicore
  - SPDS

- Dual-Redundant Controllers
  - RC&IS (includes rod server processing channel [RSPC], rod action and position information [RAPI], file control module [FCM], Signal interface unit [SIU]),
  - Automated thermal limit monitor (ATLM), and
  - Rod worth minimizer (RWM)

- Triple-Redundant Controllers
  - DPS

The PIP (Train A and Train B) includes the following:

- Control rod drive (CRD) system,
- Reactor water cleanup and shutdown cooling (RWCU/SDC) system,
- FAPCS,
- Nonsafety remote shutdown system (RSS),
- Reactor component cooling water system (RCCWS),
- Plant service water system (PSWS),

- PSWS cooling towers,
- Nuclear island chilled water system (NICWS),
- Drywell cooling nonsafety electrical systems,
- Instrument air system (IAS),
- Nonsafety PAM systems,
- Nonsafety LD&IS systems,
- PCCS ventilation fans,
- Ancillary and standby diesel generators,
- 6.9-kilovolt (kV) plant electrical power system,
- Low voltage electrical system,
- Nonsafety uninterruptible power supplies (UPS)

The BOP systems include the following:

- SB&PC
- FWCS
- Feedwater temperature control system (FWTCS)
- TGCS
- Turbine auxiliary;
- Generator auxiliary controller;
- Electrical system main transformer/unit auxiliary transformer (UAT) controller;
- Main condenser controller;
- Electrical system reserve auxiliary transformer (RAT) controller;
- Normal heat sink controller;
- Condensate/feedwater/drains/extraction controller, including extraction and level control;
- Water systems controller;
- Service air/containment inerting/floor drains controller; and
- Miscellaneous heating, ventilation, and air conditioning (HVAC) controller.

The plant computer functions group includes the following:

- Performance monitoring and control (PMC) functions, prediction calculations, visual display control, point log and alarm processing, surveillance test support, and automation;

- Core thermal power and flow calculations;

- The plant AMS that alerts the operator to process deviations and equipment and instrument malfunctions;

- Fire Protection System (FPS) data through datalinks and gateways;

- The Historian function, which stores data for later analysis and trending;

- Control of the main mimic on the MCR wide display panel (WDP);

- Support functions for printers and the secure data communications to the technical support center (TSC), emergency operation facility (EOF), Emergency Response Data System (ERDS), and potential links to the simulator;

- OLPs to guide the operator during normal and abnormal operations, and to verify and record compliance;
- Transient recording;
- Nonsafety PAM displays;
- Report generators to allow the operator, technician, or engineer to create historical or real time reports for performance analysis and maintenance activities;
- The plant configuration database (PCD) to document, manage, and configure components of the N-DCIS;
- Gateways to vendor-supplied nonsafety systems such as seismic, meteorological, and radiation monitoring; and
- Nonsafety process and area radiation monitoring

Nonsafety VDUs in the MCR and RSS panels provide the capability to display and control information regarding the plant computer functions (PCF).

The N-DCIS includes the following non segment-based equipment:

- Nonsafety VDUs/MCRP
- Gateway
- Datalinks
- SPDS logic

The N-DCIS performs control functions with logic processing modules using signals acquired by the RMUs. The N-DCIS logic is implemented in triple-redundant control systems for core nonsafety key systems such as the FWCS and the plant automation system (PAS), but it is always at least redundant for systems required for power generation. Thus, no single failure of an active DCIS component can cause or prevent a BOP trip or reactor scram.

The N-DCIS provides the control and monitoring operator interface on nonsafety VDUs in the MCR and RSS panel. The VDUs operate independently of one another, yet each can normally access any component in the N-DCIS. This gives the RSS panels the same control and monitoring capability as the displays in the MCR. The N-DCIS components that are key for power generation are provided with two or three UPSs with battery backup for at least 2 hours. For loss-of-offsite-power events or after battery backup power is lost, the N-DCIS can operate from either of the two diesel generators. The N-DCIS provides self-diagnostics that monitor communications, power, and other failures to the replacement card, module, or chassis level. Process diagnostics include system alarms in the MCR and the ability to identify sensor failures.

The nonsafety DPS is designed to mitigate the possibility of digital protection system CCF discussed in Item II.Q of SECY-93-087. The DPS is a triple-redundant system, powered by redundant nonsafety load group power sources. The DPS provides diverse reactor protection using a subset of the RPS scram signals. The DPS also provides diverse emergency core cooling by independently actuating the ECCS and selected containment isolation functions. The DPS processes the nonsafety portions of the ATWS mitigation logic.

The RC&IS provides for normal monitoring of control rod positions and executing normal control rod movement commands. The RC&IS uses a dual-redundant architecture of two independent channels. The failure or malfunction of the RC&IS has no impact on the hydraulic scram function of the CRD. The circuitry for normal insertion and withdrawal of control rods in RC&IS is completely independent of the RPS circuitry controlling the scram valves.

The FWCS is a power generation system for the purpose of maintaining proper vessel water level and feedwater temperature. The FWCS uses a triple-redundant, fault-tolerant digital controller (FTDC). The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FWCS is not safety related and is not required for safe shutdown of the plant. The FWCS initiates a runback of feedwater demand upon receipt of an ATWS trip signal from the ATWS/SLC logic.

The PAS provides reactivity control, heatup and pressurization control, reactor power control, generator power control, and plant shutdown control. The PAS consists of triple-redundant process controllers. The PAS accomplishes different phases of reactor operations, which include approach to criticality, heatup, reactor power increase, automatic load following, reactor power decrease, and shutdown. The PAS interfaces with the operator's console to perform its designed functions. In the automatic mode, the PAS issues command signals to the turbine master controller, which contains appropriate algorithms for automated sequences of turbine and related auxiliary systems. The PAS does not perform or ensure any safety function.

The SB&PC system controls reactor pressure during plant startup, power generation, and shutdown modes of operation. This is accomplished through control of the turbine control valves (TCVs) and turbine bypass valves (TBVs) such that susceptibility to reactor trip, turbine generator trip, main steam isolation, and SRV valve operation is minimized. The SB&PC system uses a triple-redundant microprocessor-based FTDC controller. With triple redundancy, the loss of one or two complete processing channels will not affect the system function.

The TGCS controls the turbine speed, load, and flow for startup and normal operations. The TGCS is a triple-redundant process control system. Only the operator can switch the turbine generator controller to Automatic (remote). The operator or the automatic power regulator can switch the turbine generator controller to Manual (local). The TGCS interfaces with the SB&PC system.

3D Monicore provides core performance information and has two major components, the monitor and the predictor. Both components use a three-dimensional core model as the main calculation engine. 3D Monicore is designed to periodically track current reactor parameters automatically with live plant data. This allows the user to study the effects of different rod patterns, core flows, and fuel burnups before performing reactor maneuvers to support plant operation.

In the event of a core-melt accident in which molten fuel reaches the lower drywell, the flow through the deluge line is required to flood the lower drywell region with a required deluge network flow rate assumed in the accident analyses. Deluge line flow is initiated by thermocouples, which sense high lower drywell region temperatures indicative of molten fuel on the lower drywell floor. Logic circuits actuate squib-type valves in the deluge lines upon detection of a drywell temperature exceeding the setpoint value, provided that another set of dedicated temperature switches also senses the drywell temperature to be higher than a preset value. The deluge logic is completely separate from and independent of the Q-DCIS and the N-DCIS and is powered by dedicated batteries supported by battery chargers operating on nonsafety power. Under a loss of power condition, including station blackout (SBO), the deluge system batteries provide power for 72 hours.

7.1.1.2.4 Distributed Control and Information System Design Features

The Q-DCIS is organized into four physically and electrically separate divisions. IEEE Std 603 serves as the design basis for the Q-DCIS.

As discussed in the probabilistic fire analysis, the Q-DCIS cabinet areas contain the control and information cabinets of the four safety divisions in the control building. Each of the four safety divisions is located in a separated fire area. These areas contain the equipment needed for the actuation of safety systems. A fire in a single fire area can only affect one safety system division.

The MCR contains no electrical system protection equipment (e.g., circuit breakers) that controls anything other than MCR equipment. A MCR fire will not actuate any DCIS controls, other than trip the main generator. A MCR fire does not result in the loss of offsite power or the loss of the diesels.

The Q-DCIS is designed in such a way that no single failure (including a single fire event) can spuriously actuate the containment isolation valves or inadvertently actuate an ADS SRV or DPV. Typically, two load drivers need to be actuated simultaneously to actuate the component. Each of the load drivers (which each use triply redundant logic) is actuated by the SSLC/ESF triply redundant logic signals, which are then sent to the load drivers/discrete output for the ADS SRV and DPV operated by that division. The load drivers/discrete outputs are wired in series for each valve, so that both are required for operation. This scheme makes the logic single-failure-proof against inadvertent actuation.

Each of the SRVs is equipped with four solenoid-operated pilot valves. Three solenoids receive a Q-DCIS signal; the fourth is part of the DPS. The solenoid-operated pilot valves are powered by a 120-volt alternating current (volt ac) UPS. The divisional safety power sources are located in the divisional battery rooms. The power and control cables are physically separated from other divisions.

The N-DCIS design bases include the following:

- Diversity where required from the RPS and SSLC/ESF
- Single-failure proof for power generation by redundant power and redundant communications
- Triple redundancy where required by high reliability systems (SB&PC, PAS, FWCS, turbine generator)
- Segmentation of PIP A, PIP B, BOP, GENE network (contains gateways and the DPS), and PCFs

Gateways are used to translate information from one platform to another. Normal gateways are used to put safety data onto the nonsafety networks for use in monitoring, alarming, and recording. Gateways are also used between different N-DCIS components. Some components do not require gateways because they are directly connected through fiber cables. Gateways are nonsafety components and are not treated as isolators between safety and nonsafety components. Gateways are designed to handle their required number of signals at their required speed. The single failure of any gateway or datalink will not cause a scram or loss of power generation.

PIP takes advantage of the network switch capability by segmenting specific DCIS functions. “A” and “B” PIP functions are segregated to different controllers. PIP-A controllers are connected to PIP-A network-managed switches. PIP-B controllers are connected to PIP-B network-managed switches. MCR displays are segregated into PIP-A, PIP-B, BOP, and network switches. Individual segments are still dual redundant.

The backbone of the N-DCIS consists of multiple network-managed switches, which are highly reliable and very fast. In summary, the N-DCIS configuration is as follows:

- Redundant
- Segmented
- Single-failure proof
- Capable of handling data rates for both transient and steady states in control, alarming, monitoring, and recording functions

7.1.1.3 *Staff Evaluation*

SRP Section 7.1 describes the procedures to be followed in reviewing any I&C system, including embedded computers and software necessary to support the operation of safety systems. All I&C systems important to safety are required to be identified in DCD Section 7.1 and discussed in subsequent sections of DCD Tier 2, Chapter 7. The safety systems supported by I&C systems are described in other sections of DCD Tier 2 (particularly in Chapters 5, 6, 8, 9, 10, 15, and 18).

This section evaluates each regulation and acceptance criterion applicable to the DCIS using SRP Appendix 7.1-A supplemented by SRP Appendixes 7.1-C and 7.1-D, which provide additional guidance for evaluating compliance with IEEE Std 603. The evaluation is intended to allow cross-referencing by specific I&C systems since the basis for the DCIS conformance with the acceptance criteria typically applies to specific I&C systems. Sections 7.2 through 7.8 of this report focus on system-specific acceptance criteria identified in the corresponding SRP sections.

7.1.1.3.1 Design Acceptance Criteria Process for Compliance with Regulations

The NRC implements the policy (SECY-92-053) of accepting the use of DAC, considered a special kind of ITAAC, in lieu of detailed design information in the digital I&C area. The applicant proposes and the staff reviews, approves, and certifies, sufficient ITAAC to ensure that the licensee will meet the DAC during construction before loading fuel. The NRC allows the use of the DAC process because providing detailed design information is not desirable for applicants using technologies that change so rapidly that the design may have become obsolete between the time the NRC certifies the design and the time a plant is eventually built. For this section and the remaining sections of this report, the use of the acronym DAC/ITAAC refers to the DAC and any associated ITAAC.

An overview of the staff's review of the digital I&C design, and how DAC are used to complete the design detail, follows. This section also describes the DAC/ITAAC associated with digital I&C design, which includes the DAC/ITAAC for the HFE design process outlined in DCD Tier 1, Section 3.3, and the ITAAC for PAM instrumentation and environmental qualification (EQ) in DCD Tier 1, Sections 3.7 and 3.8, respectively. As these other ITAAC are referenced throughout Chapter 7 of this report, a brief description of each is provided for clarity.

(1) Compliance with IEEE Std 603 (DCD Tier 1, Section 2.2.15)

The I&C system uses the distributed digital system to perform plant protection and safety monitoring functions, as well as control functions. To ensure that the digital I&C system is implemented properly, the staff considered existing regulatory requirements, guides, and standards in the SRP. The staff follows the guidance provided in SRP Chapter 7, Appendices 7.1-C and 7.1-D, to verify that DCD Tier 2, Revision 9 has addressed all the criteria listed in IEEE Std 603, as required by 10 CFR 50.55a(h)(3). In accordance with the NRC policy in SECY-92-053, the applicant has opted to use the DAC process in lieu of providing the design detail for the digital I&C system.

The staff first performed a functional review at the simplified block diagram level. Sections 7.1.1, 7.1.2, and 7.1.3, and Figure 7.1.1 of DCD Tier 2, Revision 9, provide references to applicable requirements for the design of the DCIS, which includes a simplified block diagram of the I&C system, a network diagram of the DCIS, and high-level functional requirements for the DCIS. Subsequent sections of DCD Tier 2 describe the high level system functional requirements for the DCIS, such as the reactor trip system (RTS) and ESF actuation systems. The regulatory requirements referenced in the DCD for the digital I&C system establish the design criteria related to postulated single failures, CCFs, appropriate signal isolation, and so forth. These issues are discussed in more detail throughout Chapter 7 of this report. The staff finds that the DCD adequately supports the staff's review at the simplified block diagram level.

The second part of the staff's review addressed the implementation of the digital I&C system design to meet the functional system requirements. The IEEE Std 603 criteria provide the bases for the DAC/ITAAC for the digital I&C system development process. DCD Tier 2, Revision 9, specifies conformance to IEEE Std 603 throughout Chapter 7. DCD Tier 1, Revision 9, Section 2.2.15, contains the DAC/ITAAC to confirm I&C system compliance with IEEE Std 603. DCD Tier 2, Revision 9, Section 7.1.2.4, specifies conformance to applicable RGs and industry standards, as described and evaluated in Sections 7.1.1.3.3, 7.1.1.3.4, 7.1.1.3.6, and 7.1.1.3.8 of this report. These RGs and industry standards provide more detailed guidance for implementing the design criteria in IEEE Std 603. The DAC in DCD Tier 1, Revision 9, Section 2.2.15, consist of block level failure modes and effects analyses (FMEAs) and inspections of simplified logic diagrams, system design specifications, safety analyses, piping and instrumentation diagrams (P&IDs), electrical one-line diagrams, and the project design manual. These standard I&C design practices provide objective means to verify the design. NEDE-33226P describes the hardware development process integrated into the software life cycle process, which provides a phased approach for completing the DAC/ITAAC. NEDE-33226P, Figure 5-11, provides a high-level flow of the hardware design process alongside the software development process. NEDE-33226P, Section 5.7.4, describes the hardware and software specifications produced during the requirements phase of the life cycle process. NEDE-33226P, Section 5.7.6, describes the system requirements specification which identifies additional hardware requirements. Section 7.1.2 of this report evaluates NEDE-33226P. Taken as a whole, the above constitutes an acceptable process for complying with IEEE Std 603.

The staff's Requests for Additional Information (RAIs) 7.1-9 through 7.1-30 concerned design compliance with IEEE Std 603. In response, the applicant created a new Section 2.2.15 in DCD Tier 1, Revision 4, for I&C compliance with IEEE Std 603. Section 2.2.15 contains the DAC/ITAAC for IEEE Std 603 (Table 2.2.15-2) necessary to confirm I&C system compliance with IEEE Std 603 and DCD Tier 2, Table 2.2.15-1, which identifies the applicability of functional systems to IEEE Std 603 DAC/ITAAC. The staff accepted the DAC approach to address compliance with the requirements of IEEE Std 603 but stated that DCD Tier 1, Revision 4, Section 2.2.15, addresses only certain sections of IEEE Std 603. In RAI 14.3-265, the staff asked the applicant to address all IEEE Std 603 sections in DCD Tier 1, Section 2.2.15 (thus superseding RAIs 7.1-9 through 7.1-30). For any IEEE Std 603 sections not provided with DAC/ITAAC, the staff requested that the applicant identify how compliance will be substantiated or provide links to existing non system-based ITAAC. RAI 14.3-265 was being tracked as an open item in the SER with open items. With its response to RAI 14.3-265 S01, the applicant also submitted responses to RAIs 7.1-99, 7.1-100, and 7.1-101, all of which are incorporated in DCD Revision 6. The applicant updated DCD Tier 1, Table 2.2.15-1, to include all applicable IEEE Std 603 criteria for all safety I&C systems. The DCD Tier 1, Section 2.2.15, design description identifies that some IEEE Std 603 criteria do not appear in Table 2.2.15-1 because some IEEE Std 603 criteria do not require ITAAC consistent with NRC guidance or because the criteria are covered by other non-system ITAAC. Table 2.2.15-2 identifies design commitments for each IEEE Std criterion for the software projects. The ITAAC acceptance criterion contain two phases: (a) the DAC phase, which specifies software projects design requirements and (b) the ITAAC implementation phase, which specifies methods to verify that the as-built design has satisfied the requirements of IEEE Std 603.

To support DCD Tier 1, Section 2.2.15 requirements, the applicant also updated DCD Tier 2, Table 7.1-2. Table 7.1-2 provides detailed cross-references to the DCD Tier 2 sections that describe the specific design and methods necessary to satisfy the IEEE Std 603 requirements. The staff finds that the applicant's response to 14.3-265 is acceptable since the applicant revised DCD Tier 1, Section 2.2.15, and finds that the sections referenced by DCD Tier 2, Table 7.1-2, properly address compliance with IEEE Std 603. Based on the applicant's responses, RAIs 7.1-9 through 7.1-30 and 14.3-265 are resolved.

(2) Software Development Activities (DCD Tier 1, Section 3.2)

In the software development area, the staff's acceptance of the software for safety system functions is based on (1) confirmation that acceptable plans are prepared to control software development activities, (2) evidence that the plans are followed in an acceptable software life cycle, and (3) evidence that the process produces acceptable design outputs. As discussed in Section 7.1.1.3.1 of this report, the NRC implements the policy (SECY-92-053) of accepting the use of DAC in lieu of detailed design information in the digital I&C area during the design certification stage. The staff follows the guidance provided in BTP HICB-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems," for evaluating software life cycle processes for digital computer-based I&C systems. Similar to the case of hardware, the applicant has not completed the software life cycle processes for design certification. Rather, the applicant has provided a software development process in NEDE-33226P and NEDE-33245P, which are incorporated by reference into DCD Tier 2. The staff finds that the applicant's software development process is acceptable for the following reasons:

- The applicant's software development process is based on and commits to the guidance in BTP HICB-14. This BTP provides detailed guidelines for evaluating software life cycle processes for digital computer-based I&C systems. These guidelines are based on reviews of licensee submittals, EPRI's requirements for advanced reactor designs, and the analysis of standards and practices documented in NUREG/CR–6101, "Software Reliability and Safety in Nuclear Reactor Protection Systems". The structure of this BTP is derived from the review process described in the SRP Appendix 7.0-A.

- RG 1.152, which endorses IEEE Std 7-4.3.2, is the primary guidance identified in BTP HICB-14 for complying with the requirements applicable to safety systems that use digital computer systems. However, numerous other RGs also are addressed and discussed in the various BTPs acceptance criteria sections. Additionally, while many standards exist that can be used to develop software for safety systems, the information in this BTP is generally based on the standards and RGs referred to in Section 7.1.2.1 of this report. The combination of these standards and RGs set bounding limits upon which the staff can rely to determine acceptability at the design certification stage.

- NEDE-33226P and NEDE-33245P describe the applicant's commitment to and implementation of BTP HICB-14, including the standards and RGs referenced therein. The applicant identified deviations from BTP HICB-14, which the staff evaluated in Section 7.1.2 of this report and finds it to be acceptable. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to verify that the software plans are developed and implemented consistent with the software development process and produce acceptable design outputs.

(3) Human Factors Engineering (DCD Tier 1, Section 3.3)

Several IEEE Std 603 sections relate to HFE: Section 5.8 (information displays), Section 5.14 (human factor considerations), and Sections 6.2 and 7.2 (manual control). SRP Section 7.5 identifies two additional topics related to HFE alarms and PAM instrumentation. The DAC/ITAAC for HFE in DCD Tier 1, Revision 9, Section 3.3, address IEEE Std 603, Sections 5.8 and 5.14 (DCD Tier 2, Revision 9, Sections 7.1.6.6.1.9, and 7.1.6.6.1.15, respectively). Since the design of human system interfaces has not been completed for the ESBWR, DCD Tier 2, Chapter 18 describes the applicant's HFE design processes. Chapter 18 of this report evaluates whether the applicant's HFE design processes incorporate accepted HFE practices and guidelines using the acceptance criteria in NUREG–0711, "Human Factors Engineering Program Review Model." DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for verifying the implementation the HFE design processes, which includes the topics listed above. Since the above topics have both HFE and I&C requirements and guidelines that should be addressed in an integrated manner, the DAC/ITAAC included in DCD Tier 1, Section 3.3, are referenced throughout Chapter 7 of this report.

(4) PAM Instrumentation (DCD Tier 1, Section 3.7)

Section 7.5.2.3 of this report evaluates the PAM instrumentation. The selection of accident monitoring variables is integrated with the HFE design process as described in DCD Chapter 18. DCD Tier 1, Revision 9, Section 3.7, which is referenced throughout Chapter 7 of this report, includes these criteria and the ITAAC to confirm that PAM instrumentation is installed consistently with the selected variable.

(5) Environmental and Seismic Qualification of Mechanical and Electrical Equipment (DCD Tier 1, Section 3.8)

While there are no DAC for the EQ and seismic qualification of mechanical and electrical equipment, the ITAAC for EQ and seismic qualification in DCD Tier 1, Revision 9, Section 3.8

confirm digital I&C system conformance with IEEE Std 603, Sections 5.4 and 5.5 (these criteria are described in DCD Tier 2, Revision 9, Sections 7.1.6.6.1.5 and 7.1.6.6.1.6, respectively), and are referenced throughout Chapter 7 of this report. Thus the DCD Tier 1, Section 3.8, ITAAC is associated with the DAC/ITAAC for IEEE Std 603 discussed in Item (1) above. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment. Chapter 3 of this report evaluates these DCD Tier 2 sections.

(6) DAC for Other SRP Acceptance Criteria

As described in SRP Appendix 7.1-A, compliance with the SRP acceptance criteria (e.g., certain GDC requirements) depends on compliance with some or all of the IEEE Std 603 requirements. Accordingly, the evaluation of the SRP acceptance criteria depends upon DAC for the IEEE Std 603 requirements.

7.1.1.3.2 General Conformance of the Distributed Control and Information System with Standard Review Plan Criteria

As required by 10 CFR 52.47(a)(9), the application must include an evaluation of the design against the SRP revision in effect 6 months before the docket date of the application and must identify all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, analytical techniques, and procedural measures given in the SRP acceptance criteria. SRP Table 7-1 provides a matrix identifying the regulatory requirements, acceptance criteria, and guidance and their applicability to the various sections of Chapter 7 of the safety analysis report (SAR) (DCD for design certification). DCD Tier 2, Table 7.1-1 identifies all of the applicable regulatory requirements, acceptance criteria, and guidance for each I&C system in the design. Table 7.1-2 provides detailed cross-references to the DCD Tier 2 section describing the specific design and method to satisfy IEEE Std 603 requirements. These tables assisted the staff in identifying the related documentation within the DCD necessary to address compliance with the regulatory requirements for I&C systems important to safety.

DCD Tier 2, Table 7.1-1, identifies applicable regulatory requirements and guidelines for I&C systems. The staff performed its review using SRP Table 7.1, Revision 5 and SRP Appendix 7.1-A, Revision 5. However, the applicant was required to address Revision 4, which was in effect 6 months before the docketing of the design certification. SRP Table 7.1, Revision 5 and SRP Appendix 7.1-A, Revision 5, provide an expanded list of GDC applicable to I&C systems as compared to the list provided in SRP Appendix 7.1-A, Revision 4. Revision 5 of the SRP explicitly addresses several GDC that were implicitly included in Revision 4 through the table associated with GDC 13; hence, Revision 5 does not represent a change in regulatory requirements with respect to the GDC. While the applicant did not include all of the GDC listed in SRP Table 7.1, Revision 5, in DCD Tier 2, Table 7.1-1, the staff verified that all of the applicable GDC are addressed in other parts of the DCD, as described below.

The applicant has agreed to meet the SRP guidance with a few exceptions. These exceptions are noted in DCD Tier 2, Revision 9, Sections 1.9 and 3.1, as well as the applicable sections of this report. In DCD Tier 2, Table 1.9-7, the applicant provided clarifications with respect to six BTPs. Because of the unique design features related to a passive ESBWR, BTPs HICB-2, “Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines,” HICB-3, “Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps out of Service,” HICB-4, “Guidance on Design Criteria for Auxiliary Feedwater Systems,” HICB-5,”Guidance on Spurious Withdrawals of Single Control Rods in

Pressurized Water Reactors," HICB-6, "Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode," and HICB-13, "Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors," do not apply to the design. The staff finds these clarifications acceptable.

SRP Chapter 7, Appendix 7-A, provides an agenda for the station site visit related to the I&C systems and includes a verification of layouts, separation and isolation, test features, and the potential for damage resulting from fire, flooding, or other environmental effects. The review described in SRP Chapter 7, Appendix 7-A, will be accomplished as part of the testing and inspections done by the COL licensees referencing the design certification.

7.1.1.3.3 Compliance with 10 CFR 50.55a(a)(1)

10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The staff evaluated whether the DCIS adequately addresses the requirements of 10 CFR 50.55a(a)(1) in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states that the applicant should commit to conformance with the RGs and industry standards referenced in SRP Sections 7.1 through 7.9 and the BTPs listed in SRP Appendix 7-A. The discussion of the RGs in SRP Appendix 7.1-A identifies the applicable RGs and standards. SRP Table 7.1 identifies that 10 CFR 50.55a(a)(1) applies to all I&C systems and components (all DCD Chapter 7 sections) and specifies the applicability of specific RGs to particular Chapter 7 sections.

DCD Tier 2, Revision 9, Sections 7.1.2.4 and 7.1.4.4 specify conformance to RGs and industry standards for the Q-DCIS and the N-DCIS, respectively. DCD Tier 2, Table 7.1-1 identifies the applicability of specific RGs to particular I&C systems. DCD Tier 2, Table 1.9-22 specifies the versions of the industrial code and standards applicable to the ESBWR. Using the preceding listed DCD sections, the staff verified that the I&C standards listed in the DCD are consistent with SRP Appendix 7.1-A with some limited exceptions. DCD Tier 2, Revision 9, Table 1.9-22 specifies conformance to Instrumentation, Systems, and Automation Society (ISA)-S67.04-2006, "Setpoints for Nuclear Safety-Related Instrumentation," instead of ISA-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation," for setpoint methodology. Section 7.1.4 of this report evaluates the setpoint methodology. NEDE-33226P and NEDE-33245P identify deviations from BTP HICB-14, five RGs, and associated industry standards. Section 7.1.2.3.4 of this report evaluates these deviations, which the staff finds is acceptable. DCD Tier 2, Revision 9, Table 7.1-1 excludes four RGs (RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision on Plant-Specific Changes to the Licensing Basis," RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," RG 1.189, "Fire Protection for Nuclear Power Plants," and RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities") from its applicability matrix, and DCD Tier 2, Revision 9, Table 1.9-7, identifies differences with the SRPs with regard to RGs 1.22, 1.118, and 1.151. Section 7.1.1.3.8 of this report evaluates these exclusions and differences, which the staff finds is acceptable. However, in RAI 7.1-100 the staff requested the applicant to consistently identify standards for specific systems used to ensure conformance with the IEEE Std 603 criteria. An example is provided in RAI 7.1-100, Item D. In RAI 7.1-136, the staff requested the applicant to address in DCD Tier 2, Section 7.1.6.6, the RGs and standards used to ensure conformance with the IEEE Std 603 criteria. RAIs 7.1-100 and 7.1-136 were being tracked as open items in the SER with open items. In its responses, the applicant addressed the conformance of the safety I&C designs to both RG 1.53 and IEEE Std 379-2000, "Application of the Single Failure

Criterion to Nuclear Power Generating Station Safety Systems." The applicant added a discussion about conformance to IEEE Std 379 to DCD Tier 2, Revision 9, Section 7.1.6.6, which is the basis for modifying the conformance statement for RG 1.53. DCD Tier 2, Subsections 7.1.6.4, 7.2.1.3.4, 7.2.2.3.4, 7.2.3.3.4, 7.3.1.1.3.4, 7.3.1.2.3.4, 7.3.3.3.4, 7.3.4.3.4, 7.3.5.3.4, 7.3.6.3.4, 7.4.1.3.4, 7.4.2.3.4, 7.4.4.3.4, 7.4.5.3.4, 7.5.2.3.4, 7.5.3.3.4, 7.6.1.3.3, and 7.8.3.4 consistently document the conformance statements for RG 1.53. The staff confirmed that the applicant incorporated the DCD changes into DCD Revision 8. The staff finds that the response is acceptable since the applicant properly addressed conformance to RGs and compliance with industry standards. Based on the applicant's responses, RAIs 7.1-100 and RAI 7.1-136 are resolved. Based on the above, the staff finds that 10 CFR 50.55a(a)(1) is adequately addressed for the DCIS.

7.1.1.3.4 Compliance with 10 CFR 50.34(f), 10 CFR 50.62 and 10 CFR 52.47(b)(1)

The applicant identified the following Three Mile Island (TMI) Action Plan items that do not apply to the ESBWR design because the ESBWR relies on passive plant design features and not on the active systems identified below:

- Item II.K.3.13 - high pressure coolant injection (HPCI) and reactor core isolation coolant (RCIC) initiation levels

- Item II.K.3.15 - isolation of HPCI and RCIC (turbine-driven)

- Item II.K.3.21 - automatic restart of low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI)

- Item II.K.3.22 - RCIC automatic switchover of suction supply

The staff agrees with the applicant's determination.

The TMI Action Plan items described below apply to the ESBWR design. (The applicable regulation is identified followed by the TMI Action Plan Item number in brackets)

(1) 10 CFR 50.34(f)(2)(v) [Item I.D.3], Bypass and Inoperable Status Indication

The staff evaluated whether the applicant adequately addressed 10 CFR 50.34(f)(2)(v) [Item I.D.3] for the DCIS. According to SRP Table 7-1 and SRP Appendix 7.1-A, 10 CFR 50.34(f)(2)(v) [Item I.D.3] applies to the protection systems (RTS and ESF), information systems important to safety, interlock logic, and supporting systems (DCD Sections 7.2, 7.3, 7.5, and 7.6). Furthermore, 10 CFR 50.34(f)(2)(v) [Item I.D.3] requires an applicant to provide an automatic indication of the bypassed and operable status of the safety systems. DCD Tier 2, Revision 9, Table 7.1-1, identifies 10 CFR 50.34(f)(2)(v) [Item I.D.3] as being applicable to the safety systems consistent with SRP Table 7-1. DCD Tier 2, Revision 9, Table 1A-1, states that the I&C design provides an automatic indication of the bypasses and inoperable status of safety systems. This table also identifies where the applicability of 10 CFR 50.34(f)(2)(v) is discussed in DCD Sections 7.2, 7.3, 7.5, and 7.8.

SRP Section 7.5 states that the acceptance criteria for bypass and inoperable status indication are addressed in part by conformance to RG 1.47. SRP Table 7-1 and SRP Appendix 7.1-A states that RG 1.47 applies to the same systems as 10 CFR 50.34(f)(2)(v) [Item I.D.3]. DCD Tier 2, Revision 9, Table 7.1-1, identifies that RG 1.47 applies to the safety systems consistent with SRP Table 7-1. In a discussion of the conformance of the DCIS to RG 1.47, DCD Tier 2, Revision 9, Section 7.1.6.4, states that bypass indications are designed to satisfy the guidance of IEEE Std 603, Section 5.8.3 and RG 1.47. This section also states that bypass indications use isolation devices that preclude the possibility of any adverse electrical effect of the bypass indication circuits on the plant's safety systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.9, addresses the information display criterion (IEEE Std 603, Section 5.8) that requires (1) information displays for the referencing platform be designed to be accessible to the operators, (2) display variables for manually controlled actions, (3) display system status information, (4) indication of bypasses, and (5) PAM variables be displayed in accordance with the HFE design process. DCD Tier 1, Revision 9, Section 3.3, includes DAC/ITAAC for the HFE design process. Based on the above and the inclusion of RG 1.47 in the applicable safety system design bases, the staff finds that conformance with the guidance in RG 1.47 is adequately addressed.

As discussed in SRP Appendix 7.1-A, the provisions of 10 CFR 50.34(f)(2)(v) [Item I.D.3] may be addressed by conformance with IEEE Std 603, Sections 5.6, 5.8, 5.12, and 6.3. Section 7.1.1.3.10 of this report presents the staff's evaluation of IEEE Std 603, Sections 5.6, 5.8, 5.12, and 6.3. The staff finds that Sections 5.6, 5.8, 5.12, and 6.3 of IEEE Std 603 are adequately addressed, based on their inclusion in the safety systems design bases and in DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 sections. Information displays are designed using the HFE design process, as described in DCD Tier 2, Revision 9, Chapter 18 and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification applies to all safety systems and includes bypasses and inoperable status indications. Accordingly, based on the inclusion of 10 CFR 50.34(f)(2)(v) [Item I.D.3] and associated criteria and guidelines in the safety systems' design basis and their confirmation in the DAC/ITAAC, the staff finds that the requirements of 10 CFR 50.34(f)(2)(v) [Item I.D.3] are adequately addressed for the DCIS.

(2) 10 CFR 50.34(f)(2)(xi) [Item II.D.3], Direct Indication of Relief and Safety Valve Position

10 CFR 50.34(f)(2)(xi) [Item II.D.3] requires an applicant to provide direct indication of relief and safety valve positions (open or closed) in the control room.

DCD Tier 2, Revision 9, Table 1A.1, specifies that a direct indication of SRV and DPV positions (open or closed) be provided in the MCR. DCD Tier 2, Revision 9, Section 7.3.1.1.5, also specifies that the ADS I&C indicates the status of the SRV and DPV in the MCR in conformance with IEEE Std 603, Section 5.8. DCD Tier 2, Revision 9, Section 7.1.6.6.1.9 identifies that information display design is part of the HFE design process as described in DCD Tier 2, Revision 9, Chapter 18 and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification is applicable to all safety systems and includes verifying the inventory of displays for system status indications. Accordingly, because the ADS I&C design basis includes the display of the status of the SRV and DPV and the implementation of the ADS I&C design basis is confirmed by the DAC/ITAAC, the staff finds that the requirements of 10 CFR 50.34(f)(2)(xi) [Item II.D.3] are adequately addressed for the DCIS.

(3) 10 CFR 50.34(f)(2)(xvii) [Item II.F.1], Accident Monitoring Instrumentation

10 CFR 50.34(f)(2)(xvii) [Item II.F.1] requires the applicant to provide instrumentation to measure, record, and read out in the control room (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential accident release points. The applicant must also provide for continuous sampling of radioactive iodine and particulates in gaseous effluents from all potential accident release points and to provide for an onsite capability to analyze and measure these samples.

In DCD Tier 2, Revision 9, Section 7.5.2, the applicant stated that the CMS provides the instrumentation to monitor the following:

- The atmosphere in the containment for high gross gamma radiation levels
- The pressure of the drywell and wetwell
- The drywell/wetwell differential pressure
- The lower and upper drywell water level (post-LOCA)
- The temperature of the suppression pool water
- The suppression pool water level
- The drywell/wetwell hydrogen/oxygen concentration
- The containment area radiation

These parameters are monitored during both normal reactor operations and post-accident conditions to evaluate the integrity and safe conditions of the containment. Abnormal measurements and indications initiate alarms in the MCR.

DCD Tier 2, Revision 9, Section 7.5.1, describes the PAM instrumentation, including the process to identify the post-accident plant parameters to be displayed in the MCR. The PAM instrumentation has the following safety design basis:

- Provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions, as appropriate, to ensure adequate safety.

- Provide the appropriate MCR instrumentation and displays to provide the information from which actions are taken to maintain a safe plant condition under accident conditions, including LOCAs.

- Provide equipment (including the necessary instrumentation) at appropriate locations outside the MCR with the capability for prompt hot shutdown of the reactor.

- Provide the means for monitoring the reactor containment atmosphere, spaces containing components that recirculate LOCA fluids, effluent discharge paths, and plant environs for radioactivity that may be released as a result of accidents.

As discussed above, the PAM instrumentation and the CMS provide the accident monitoring instrumentation functions required by 10 CFR 50.34(f)(2)(xvii) [Item II.F.1]. As described in Section 7.5 of this report, the acceptability of the PAM instrumentation and the CMS and their conformance to 10 CFR 50.34(f)(2)(xvii) [Item II.F.1] depend on (1) the inclusion of the accident monitoring instrumentation functions in the systems design bases; (2) the inclusion of IEEE Std 603 criteria in the systems design bases; (3) DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria; (4) the description in the DCD of the performance-based criteria the ESBWR uses for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables consistent with RG 1.97 and IEEE Std 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations;" (5) DCD Tier 1, Revision 9, Section 3.7, including the performance-based criteria in the ITAAC to confirm that the PAM instrumentation is installed consistent with the selected variables; and (6) DCD Tier 1, Revision 9, Section 3.3, including the DAC/ITAAC to confirm that the HFE design is implemented based on the process described in DCD Chapter 18. Section 7.5 of this report evaluates Items (1) through (5), which the staff finds acceptable. DCD Tier 1, Revision 9, Section 3.7 specifies that the scope of instrumentation relied upon to fulfill the PAM function is determined through the HFE design process. For each variable and type, the process determines additional characteristics appropriate to that variable based on the guidelines provided in RG 1.97. PAM instrumentation software is developed in accordance with the software development program described in DCD Tier 1, Revision 9, Section 3.2. Based on the review of the DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.7 DAC/ITAAC documentation, the staff finds that the requirements of 10 CFR 50.34(f)(2)(xvii) [Item II.F.1] are adequately addressed for the DCIS.

(4) 10 CFR 50.34(f)(2)(xviii) [Item II.F.2], Inadequate Core Cooling Instrumentation

10 CFR 50.34(f)(2)(xviii) [Item II.F.2] requires an applicant to have instruments that provide, in the control room, an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in pressurized-water reactors (PWRs) and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and boiling-water reactors (BWRs).

To address the requirement, SRP Appendix 7.1-A states that instrumentation for the detection of inadequate core cooling should provide the operator with sufficient information during accident situations to take planned manual actions and to determine whether safety systems are operating properly. In addition, the instrumentation should provide sufficient data for the operator to be able to evaluate the potential for core uncovery and gross breach of protective barriers, including the resultant release of radioactivity to the environment.

For the design, the RPV water level is the only issue to be considered, because BWRs operate at saturation pressure and saturation monitors are not required. The detection of conditions indicative of inadequate core cooling in the design is provided by the direct RPV water level instrumentation. The RPV water level is measured by four physically separate level (differential pressure) transmitters mounted on separate divisional local racks in the safety envelope within the reactor building. Each transmitter is on a separate pair of instrument lines and is associated with a separate RPS electrical division. The instruments for monitoring the RPV water level from the sensor to the control room display are classified as safety instrumentation. Water level measurements include fuel zone, wide range, narrow range, and shutdown range. Each division has its own set of RPV sensing line nozzle connections.

The staff reviewed the RPV water-level measurement in the design. DCD Tier 2, Revision 9, Table 1A-1, identifies that the RPV water level instrumentation system design includes a constant metered addition of purge water from the CRD hydraulic system to prevent the build-up of dissolved gasses in the fixed leg. DCD Tier 2, Revision 9, Section 7.7.1.2.2, identifies that the CRD Hydraulic Subsystem provides a purge flow that keeps the RPV water level reference leg instrument lines full. These lines are filled to address the effects of noncondensable gases in the instrument lines and to prevent erroneous reference information after a rapid RPV depressurization event. DCD Tier 2, Revision 9, Section 7.7.1.3.3, states that the instrument sensing lines for the NBS conform to the guidelines in RG 1.151 and the associated guidance in ISA-S67.02.01, “Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants.”

DCD Tier 2, Revision 9, Section 7.7.1.4, identifies that water level instruments are located outside the drywell so that calibration and test signals can be applied during reactor operation, in conformance with IEEE Std 603, Section 5.7. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for the applicant to verify that maintenance bypasses are implemented in the software development process to provide the capability for the testing and calibration of the safety systems. This provision will verify the ability of RPV level measurements to accomplish their safety functions. Based on the applicant’s documentation of the TMI Action Plan item and DCD Tier 1 and Tier 2 updates to Revision 9, the staff considers that the RPV water-level measurement issues are resolved.

DCD Tier 2, Revision 9, Section 7.7.1.2.2, describes the measurement of reactor coolant temperatures. The reactor coolant temperatures are measured at the mid-vessel inlet to the RWCU/SDC system and at the bottom head drain. Coolant temperatures can also be determined in the steam-filled parts of the RPV and steam-water mixture by measuring the reactor pressure. In the saturated system, reactor pressure connotes saturation temperature. Coolant temperatures (core inlet temperature) can normally be measured by the redundant core inlet temperature sensors located in each LPRM assembly below the core plate elevation. The RPV outside surface temperature is measured at the head flange and at the bottom head locations. Temperatures needed for operation and for compliance with the TS operating limits are obtained from these measurements. The staff finds this approach acceptable.

Accordingly, based on the RPV water level and the reactor coolant temperature measurement instrumentation included in the NBS I&C design basis and its confirmation in the DCD Tier 1 Revision 9 DAC/ITAAC, the staff finds that requirements of 10 CFR 50.34(f)(2)(xviii) [Item II.F.2] are adequately addressed for the DCIS.

(5) 10 CFR 50.34(f)(2)(xiv) [Item II.E.4.2], Containment Isolation Systems

10 CFR 50.34(f)(2)(xiv) [Item II.E.4.2] requires an applicant to provide containment isolation systems that do the following:

- Ensure that all non essential systems are isolated automatically by the containment isolation system,
- For each non essential penetration (except instrument lines), have two isolation barriers in series,
- Do not result in reopening the containment isolation valves on resetting of the isolation signal,
- Utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal operation,
- Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

As described in DCD Tier 2, Revision 9, Section 7.3.3, the LD&IS is used to detect and monitor leakage from the RCPB and to initiate the appropriate safety action to isolate the source of the leak. The system is designed to automatically initiate the isolation of certain designated process lines penetrating the containment to prevent the release of radioactive material from the RCPB. DCD Tier 2, Revision 9, Table 5.2-6, identifies the fluid lines designated for closure for each monitored variable. DCD Tier 2, Revision 9, Section 6.2.4, describes the containment isolation functions and specifies that DCD Tier 2, Revision 9, Tables 6.2-15 through 6.2-45, provide information on containment isolation valves. Section 6.2.4 of this report provides the evaluation of the second bullet above concerning two isolation barriers in series.

DCD Tier 2, Revision 9, Section 5.2.5, identifies that diverse signals are provided for the containment isolation function. The signals, high drywell pressure, low reactor water level (Level 2), and the backup reactor water level (Level 1), are included in the list of monitored variables in DCD Tier 2, Revision 9, Table 5.2-6, and in the list of sensor parameters in DCD Tier 2, Revision 9, Table 7.3-5. The LD&IS functions are performed in two separate safety platforms. The MSIV isolation logic functions are performed in the RTIF-NMS platform, while all other containment isolation logic functions are performed in the SSLC/ESF platform.

DCD Tier 2, Revision 9, Section 7.3.3.3, identifies that the LD&IS logic is designed to seal-in the isolation signal once the trip is initiated. The isolation signal overrides any control action to trigger the opening of isolation valves. Reset of the isolation logic is required before any isolation valve can be opened manually. Manual valve override capability is provided for valves that are required to operate following a design basis event on a valve-by-valve or line-by-line basis. The valve override requires at least two deliberate operator actions and is under administrative controls. The override status is alarmed in the MCR. The staff finds this acceptable.

DCD Tier 2, Revision 9, Table 1A-1, in the discussion regarding 10 CFR 50.34(f)(2)(xiv), identifies that the alarm and initiation setpoints of the LD&IS are set to initiate containment isolation at the minimum values compatible with normal operating conditions for containment penetrations containing process lines that are not required for emergency operation. The value for this setpoint is based on the analytical limit used in safety analyses. The staff finds this acceptable.

DCD Tier 2, Revision 9, Table 5.2-6, identifies that the pathways to the environs, including the containment purge lines and valves and the reactor building HVAC exhaust, isolate upon containment isolation signals, refueling area air exhaust high radiation signal, and the reactor building exhaust high radiation signal. The staff finds this acceptable.

As discussed above, the LD&IS provides the containment isolation I&C functions associated with 10 CFR 50.34(f)(2)(xiv) [Item II.E.4.2]. As described in Section 7.3 of this report, the acceptability of the LD&IS and its conformance to 10 CFR 50.34(f)(2)(xiv) [Item II.E.4.2] depend on (a) the inclusion of the containment isolation I&C functions in the system design bases; (b) the inclusion of IEEE Std 603 criteria in the systems design bases; (c) DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria; (d) DCD Tier 1, Revision 9, Section 3.2, including the DAC/ITAAC to verify the implementation of the software development process; and (e) DCD Tier 1, Revision 9, Section 3.3, including the DAC/ITAAC to confirm that the HFE design is implemented based on the process described in DCD Tier 2, Revision 9, Chapter 18. Based on the review of the DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, and 3.3 DAC/ITAAC documentation and DCD Tier 2, Revision 9, Sections 7.2 and 7.3 design descriptions, the staff finds that the requirements of 10 CFR 50.34(f)(2)(xiv) [Item II.E.4.2] are adequately addressed for the DCIS.

(6) 10 CFR 50.34(f)(2)(xix) [Item II.F.3], Instruments for Monitoring Plant Conditions Following Core Damage

10 CFR 50.34(f)(2)(xix) [Item II.F.3] requires an applicant to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. DCD Tier 2, Revision 9, Table 7.1-1 identifies that PAM, CMS, PRMS, and ARMS support conformance to this requirement. The monitoring of plant conditions following core damage is a subset of the PAM functions provided by these systems. In addition, DCD Tier 2, Revision 9, Section 7.3.1.2.2, describes the design of the deluge system. The deluge system is designed to flood the containment floor in the event of a core breach that results in molten fuel on the containment floor. This system is made up of two individual and identical trains, both of which contain an automatic actuation and a manual actuation ability. There are 12 deluge valves, each with four squib initiators (each train has a manual and automatic initiator). Each of these valves feeds the Basemat-Internal Melt Arrest Coolability (BiMAC) system, which floods the containment floor following a severe accident. The logic for the deluge valves is executed in a pair of dedicated nonsafety PLCs and a pair of dedicated safety temperature switches. Automatic actuation of the deluge valves is accomplished in concert with a lower drywell high temperature. The containment floor is divided into 30 equal-area cells, with two thermocouples installed in each cell. One thermocouple from each cell is monitored in one PLC, while the other thermocouple from each cell is monitored in a second PLC. When temperatures exceed the setpoint at one set of thermocouples, coincident with setpoints being exceeded at a second set of thermocouples in adjacent cells, a trip signal is generated in each PLC.

As discussed above, the PAM, CMS, PRMS, and ARMS provide the PAM functions associated with 10 CFR 50.34(f)(2)(xix) [Item II.F.3]. As described in Section 7.5 of this report, acceptability of the PAM, CMS, PRMS, and ARMS and their conformance to 10 CFR 50.34(f)(2)(xix) [Item II.F.3], depends on (a) the inclusion of the PAM functions in the systems design bases; (b) the inclusion of IEEE Std 603 criteria in the systems design bases; (c) DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria; (d) the description in the DCD of the performance-based criteria the ESBWR uses for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables consistent with RG 1.97 and IEEE Std 497; (e) DCD Tier 1, Revision 9, Section 3.7, including the DAC/ITAAC to confirm that the PAM instrumentation is installed consistent with the selected variables; and (f) DCD Tier 1, Revision 9, Section 3.3, including the DAC/ITAAC to confirm that the HFE design is implemented based on the process described in DCD Tier 2, Revision 9, Chapter 18. Based on the review of the DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.7, DAC/ITAAC documentation, the staff finds that the requirements of 10 CFR 50.34(f)(2)(xix) [Item II.F.3] are adequately addressed for the DCIS.

(7) 10 CFR 50.34(f)(1)(vii) [Item II.K.3.18], ADS Modification That Would Eliminate the Need for Manual Activation

10 CFR 50.34(f)(1)(vii) [Item II.K.3.18] requires an applicant to perform a feasibility and risk assessment study to determine the optimum ADS design modification that would eliminate the need for manual activation to ensure adequate core cooling.

DCD Tier 2, Revision 9, Section 7.3.1.1, describes the ADS. In the design, the ECCS provides emergency core cooling in response to events that threaten reactor coolant inventory. The

ECCS comprises the ADS, the GDCS, the ICS, and the SLC system. The ADS resides within the NBS and comprises the SRVs and DPVs and associated I&C. The ADS actuation logic is implemented in four SSLC/ESF divisions, each of which can make a Level 1 trip vote. Each of the divisional trip votes is shared with the other divisions. Normally, each of the four divisions makes a two-out-of-four (2/4) trip decision from the four divisional votes. Each division of the SSLC/ESF uses triply redundant processors to implement the 2/4 trip logic that actuates three series load drivers (for DPV) or two series load drivers (for SRV solenoids) to support the requirement that single divisional failures cannot result in the inadvertent opening of any ADS valve (SRV or DPV). The ADS depressurizes the reactor to allow the low-head GDCS to provide makeup coolant to the reactor. The ADS logic resides on the SSLC/ESF portion of the Q-DCIS. The GDCS provides emergency core cooling once the reactor is depressurized. The GDCS is capable of injecting large volumes of water into the RPV to keep the core covered for at least 72 hours following a LOCA.

As discussed above, the ADS and the GDCS provide the automatic ADS I&C functions associated with 10 CFR 50.34(f)(1)(vii) [Item II.K.3.18]. As described in Section 7.3 of this report, acceptability of the ADS and the GDCS and their conformance to 10 CFR 50.34(f)(1)(vii) [Item II.K.3.18] depend on (1) the inclusion of the automatic ADS I&C functions in the systems design bases; (2) the inclusion of IEEE Std 603 criteria in the systems design bases; (3) DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria; (4) DCD Tier 1, Revision 9, Section 3.2, including the DAC/ITAAC to verify the implementation of the software development process; and (5) DCD Tier 1, Revision 9, Section 3.3, including the DAC/ITAAC to confirm that the HFE design is implemented based on the process described in DCD Tier 2, Revision 9, Chapter 18. Based on the review of the DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, and 3.3, DAC/ITAAC documentation, the staff finds that the requirements of 10 CFR 50.34(f)(1)(vii) [Item II.K.3.18] are adequately addressed for the DCIS.

(8) 10 CFR 50.34(f)(2)(xxiv) [Item II.K.3.23], Central Reactor Vessel Water Level Recording

10 CFR 50.34(f)(2)(xxiv) [Item II.K.3.23] requires an applicant to provide the capability to record the reactor vessel water level in one location on recorders that meet normal post accident recording requirements.

DCD Tier 2, Revision 9, Table 1A-1, shows that the recording of water levels is included in the MCR. Water level measurements are made by the wide-range and fuel-range water level instruments.

DCD Tier 2, Revision 9, Section 7.5.1.3.4, includes a description of the process, with associated performance criteria, used to develop the PAM instrumentation. The description includes a list of the performance criteria with associated variables used to develop the PAM instrumentation. A recording requirement is included as a variable for the "Display Criteria."

As discussed above, the PAM instrumentation provides the recording capability associated with 10 CFR 50.34(f)(2)(xxiv) [Item II.K.3.23]. As described in Section 7.5 of this report, acceptability of the PAM instrumentation and its conformance to 10 CFR 50.34(f)(2)(xxiv) [Item II.K.3.23], depend on (a) the inclusion of the recording capability in the system design bases; (b) the inclusion of applicable IEEE Std 603 criteria in the systems design bases; (c) DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria; (d) the description of the performance-based criteria the ESBWR uses for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables consistent with RG 1.97 and IEEE Std 497; (e) DCD Tier 1, Revision 9, Section 3.7, including the criteria and ITAAC to confirm that the PAM instrumentation is installed consistent with the selected variables, and (f) DCD Tier 1, Revision 9, Section 3.3, including the DAC/ITAAC to confirm that the HFE design is implemented based on the process described in DCD Tier 2, Revision 9, Chapter 18. Based on the review of the DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.7, DAC/ITAAC documentation, the staff finds that the requirements of 10 CFR 50.34(f)(2)(xxiv) [Item II.K.3.23] are adequately addressed for the DCIS.

(9) 10 CFR 50.34(f)(2)(xxiii) [Item II.K.2.10], Anticipatory Reactor Trip

10 CFR 50.34(f)(2)(xxiii) [Item II.K.2.10] requires, as part of RPS, an anticipatory reactor trip that would be actuated on a loss of main feedwater or on a turbine trip.

While 10 CFR 50.34(f)(2)(xxiii) [Item II.K.2.10] states that it applies only to Babcock & Wilcox (B&W) plants, it is identified as applicable to the ESBWR on a generic issues basis since the ESBWR has an anticipatory trip. DCD Tier 2, Revision 9, Section 7.2.1.2.4.2 includes anticipatory reactor trips that would be actuated on a loss of main feedwater or on a turbine trip. DCD Tier 2, Revision 9, Section 7.2.1.2.4.2, designates these, “Power generation bus loss (Loss of feedwater flow)(Run mode only)” and “Turbine stop valve (TSV) closure,” respectively. Corresponding trips are designated as initiators (as designated in DCD Tier 1) and are included in DCD Tier 1, Revision 9, Table 2.2.7-2. Based on the inclusion of the identified trips in the RPS design, the staff finds that the requirements of 10 CFR 50.34(f)(2)(xxiii) [Item II.K.2.10] are adequately addressed for the DCIS.

(10) 10 CFR 50.62, Requirements for Reduction of Risk from ATWS

The staff evaluated whether 10 CFR 50.62 is adequately addressed for the DCIS. 10 CFR 50.62 requires that BWR plants have (a1) an alternate rod insertion (ARI) system that is diverse from the RPS and the ARI must be designed to perform its function in a reliable manner and be independent from the RPS [10 CFR 50.62(c)(3)], (b) an SLC system whose initiation must be automatic and which must be designed to perform its function in a reliable manner [10 CFR 50.62(c)(4)], and (c) an automatic recirculation pump trip (RPT) [10 CFR 50.62(c)(5)]. The requirements of 10 CFR 50.62(c)(5) do not apply to the ESBWR because the design does not have a recirculation pump. SRP Table 7.1 identifies that 10 CFR 50.62 applies to the DPS (DCD Section 7.8). DCD Tier 2, Revision 9, Table 7.1-1, identifies that 10 CFR 50.62 applies to relevant safety and nonsafety I&C systems.

Section 7.8 of this report provides a detailed evaluation of 10 CFR 50.62. As described in Section 7.8 of this report, the diverse ATWS mitigation logic includes the ARI functions required by 10 CFR 50.62(c)(3), the ATWS mitigation logic includes the SLC functions required by 10 CFR 50.62(c)(4), and both of the ATWS mitigation logics have acceptable diversity from the RPS and provide reasonable assurance of functioning in a reliable manner. Accordingly, the staff finds that 10 CFR 50.62 is adequately addressed for the DCIS.

(11) 10 CFR 52.47(b)(1), ITAAC for Standard Design Certification

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are addressed for DCIS I&C systems. This regulation requires that the design certification application contain 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. The staff reviewed DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.7, DAC/ITAAC documentation, and the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the DCIS. Section 14.3 of this report provides details of the staff evaluation.

7.1.1.3.5 Resolution of Generic Issues Related to Instrumentation and Controls

(1) Generic Issue A-19, "Digital Computer Protection Systems"

Generic Issue A-19 was raised in 1978. NUREG–0933, "Resolution of Generic Safety Issues," issued August 2008, Table 2, identifies A-19 as a licensing issue, not a generic safety issue. In Generic Issue A-19, the staff identified a need to standardize the safety review of the RPS incorporating digital computers. Since 1978, the NRC has developed SRP Chapter 7 which includes the following:

- SRP Appendix 7.0-A
- SRP Appendix 7.1-A
- SRP Appendix 7.1-C
- SRP Appendix 7.1-D
- BTP HICB-14
- BTP HICB-19, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems"
- BTP HICB-21, "Guidance on Digital Computer Real-Time Performance"
- SRP Section 14.3.5

The DCD addresses conformance to the guidance listed above. Throughout Chapter 7 of this report, the staff documents its evaluation of the conformance of the design to the guidance. The staff finds that Generic Issue A-19 is adequately addressed.

(2) Generic Issue A-34, "Instruments for Monitoring Radiation and Process Variables during Accidents"

Generic Issue A-34 was initiated to develop criteria and guidelines to be used by applicants, licensees, and staff reviewers to support the implementation of RG 1.97. In 1980, the staff decided that Generic Issue A-34 was resolved and that the implementation of this item would be carried out under TMI Action Plan Item II.F.3, which is discussed in Section 7.1.1.3.4, Item (6). Accordingly, the staff finds that Generic Issue A-34 is adequately addressed.

(3) Generic Issue A-47, "Safety Implications of Control Systems"

Generic Issue A-47 identified the need to perform an in-depth review of the nonsafety control systems and to assess the effect of control system failures on plant safety. To this end, tasks were established to identify potential control system failures that, either singly or in selected combinations, could cause overpressure, overcooling, overheating, overfilling, or reactivity events.

DCD Tier 2, Revision 9, Section 7.7, addresses I&C systems for normal plant operation that do not perform plant safety functions. However, these systems do control plant processes that can have an impact on plant safety. These systems can affect the performance of safety functions either through normal operation or through inadvertent operation. Consistent with SRP Section 7.7, Section 7.7 of this report documents the staff confirmation that the failure of the control systems themselves or as a consequence of supporting system failures, such as loss of power sources, does not result in plant conditions more severe than those described in the analysis of design basis accidents and AOOs.

A specific example raised in A-47 was the automatic reactor vessel overfill protection. The trip logic that scrams the reactor at RPV water level 8 resides in the RPS and diversely at level 9 in the DPS. The FWCS and DPS reduce feed pump flow to the vessel to zero and then trip the feedwater pumps at RPV water levels 8 (FWCS) and 9 (DPS). This example illustrates that a failure of the control system does not result in plant conditions more severe than those described in the analysis of the design basis accidents and AOOs. The standard plant TS (DCD Tier 2, , Revision 9, Chapter 16) provide surveillance requirements for the RPV water Level 8 function of the RPS instrumentation.

Based on the above, the staff finds that Generic Issue A-47 is adequately addressed.

(4) New Generic Issue 45, "Inoperability of Instrumentation Due to Extreme Cold Weather"

New Generic Issue 45 involves ensuring that safety process, instrument, and sampling lines do not freeze during extreme cold weather. In response to this issue, RG 1.151 includes the acceptance criteria for the design of protective measures against freezing in instrument lines of safety systems. DCD Tier 2, Revision 9, Section 7.1.6.4, states that instrument sensing lines are designed to conform to the guidance in RG 1.151, Revision 1. Section 7.1.6.6.1.5 states that safety components are designed to be qualified to operate in the normal and abnormal environments (including temperature, humidity, pressure, radiation, seismic, and electromagnetic interference [EMI] conditions) in which they are located; therefore, inoperability of instrumentation due to extreme cold would only be applicable to the instrument sensing lines.

DCD Tier 1, Revision 9, Section 3.8 ITAAC covers the EQ program. Safety-related equipment can perform its safety function under normal, abnormal, and design basis accident conditions.

Because DCD Tier 2 specifies that the design conforms to RG 1.151, the staff finds that New Generic Issue 45 is adequately addressed.

(5) New Generic Issue 64, "Identification of Protection System Instrument Sensing Lines"

New Generic Issue 64 involves identifying the protection system equipment that is part of the protection system subject to the regulations. The staff decided that RG 1.151 and ISA-S67.02.01 address this issue.

DCD Tier 2, Revision 9, Section 7.1.6.4, states that instrument sensing lines are designed to conform to the guidance in RG 1.151, Revision 1. ISA-S67.02.01, Section 5.3, “Identification and Channel Coding,” states that the instrument sensing tubing or piping runs pertaining to safety instrument channels shall be identified and coded so as to identify its channel. Each instrument sensing line and associated valves in this channel shall have an identification tag showing the channel and unique line or valve identification number.

The identification of safety equipment is addressed by IEEE Std 603, Section 5.11, which is evaluated in section 7.1.1.3.10 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for IEEE Std 603, Section 5.11, to verify that distinct identification of each redundant portion of safety systems.

Because DCD Tier 2 specifies that the design conforms to RG 1.151, the staff finds that New Generic Issue 64 is adequately addressed.

(6) New Generic Issue 142, “Leakage through Electrical Isolators in Instrumentation Circuits”

New Generic Issue 142 involves the concern that isolation devices subjected to failure voltages or currents at less than maximum credible fault levels passed significant levels of voltage or current, but the same devices performed acceptably at maximum credible levels. The safety system on the Class 1E side of the isolation device may be affected by the passage of small levels of electrical energy, depending on the design and function of the safety system.

DCD Tier 2, Revision 9, Section 7.1.3.3, describes the interfaces between electrical divisions for logic voting, between divisional and non divisional circuits for annunciations, and so on. However, these interfaces are accomplished through a fiber optic medium that is non conductive and thus provides full safety isolation. No interlocking is provided, nor required, for these interfaces. The electrical hardware is not affected significantly by noise because of the combination of digital transmission and fiber optics incorporated in the design.

The electrical isolation of safety equipment is addressed by IEEE Std 603, Section 5.6, which is evaluated in section 7.1.1.3.10 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for IEEE Std 603, Section 5.6, to verify the applicable independence of safety systems, which includes electrical isolation.

Based on the review of DCD Revision 9, Tier 2 information, the staff finds that New Generic Issue 142 is adequately addressed.

(7) New Generic Issue 67.3.3, “Steam Generator, Staff Actions-Improved Accident Monitoring”

Lessons from a January 1982 steam generator tube rupture event resulted in New Generic Issue 67.3.3. Since the ESBWR does not have steam generators, New Generic Issue 67 is not applicable to the design. However, one recommendation, New Generic Issue 67.3.3 Item, “Improved Accident Monitoring,” applies to both PWRs and BWRs. New Generic Issue 67.3.3 involves addressing accident monitoring weaknesses by fully implementing RG 1.97. DCD Tier 2, Revision 9, Table 1.11-1, states that New Generic Issue 67.3.3 is addressed by conformance with RG 1.97, as described in DCD Tier 2, Revision 9, Section 7.5. Section 7.5 of this report presents the staff’s evaluation of the conformance of PAM instrumentation with RG 1.97, which the staff finds acceptable.

DCD Tier 1, Revision 9, Section 3.7 provides ITAAC to confirm that the installed PAM instrumentation conforms to the guidance of RG 1.97.

The staff finds this treatment of New Generic Issue 67.3.3 acceptable. Accordingly, New Generic Issue 67.3.3 is adequately addressed.

(8) New Generic Issue 120, "On-Line Testability of Protection Systems"

New Generic Issue 120 addresses requirements for conducting at-power testing of safety system components without impairing plant operation. The staff raised this issue because it found, in the review of several plant TS in 1985, that some older plants did not provide as complete a degree of on-line testing as other plants. GDC 21 includes the requirements for on-line testing of protection systems. These requirements apply to both the RPS and the ESF actuation systems. A protection system with two-out-of-four logic that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three logic configuration meets this requirement. This issue was resolved with no new requirements.

RG 1.22 and RG 1.118, as well as IEEE Std 338-1987, "Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems," provide guidance for this issue. Conformance to these documents ensures that the ESBWR protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is at power without adversely affecting plant operation.

DCD Tier 2, Revision 9, Section 7.1.6.6.1 states that the ESBWR protection system has a two-out-of-four logic configuration that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three logic configuration. This meets the requirements in GDC 21 for on-line testing. The ESBWR design provision for testing of the protection system conforms to the guidelines in RGs 1.22 and 1.118

The on-line testability of protection systems is addressed by IEEE Std 603, Sections 5.7 and 6.5, which are evaluated in section 7.1.1.3.10 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for IEEE Std 603, Sections 5.7 and 6.5, to verify that maintenance bypasses allow test and calibration of one out of four divisions, that the divisions not in bypass status will accomplish their safety functions, that bypassed divisions alarm in the MCR, and that the division logic automatically becomes a two-out-of-three voting scheme.

The staff finds this treatment of New Generic Issue 120 acceptable. Accordingly, New Generic Issue 120 is adequately addressed.

(9) NRC Office of Inspection and Enforcement (IE) Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls"

IE Bulletin 80-06, dated March 13, 1980, requires all operating plant licensees to review plant design drawings at the schematic or elementary diagram level to determine whether, upon the reset of an ESF actuation signal, all associated safety equipment remains in its emergency mode.

DCD Tier 2, Revision 9, Section 7.3.1.1.2, calls for safety VDUs in the MCR to provide a display format allowing the operator to manually open each SRV and each DPV independently, using the primary SSLC/ESF logic function (IEEE Std 603, Sections 5.8, 6.2, and 7.2). Each nonsafety VDU in the MCR provides a display format allowing the operator to manually open each SRV independently, using the DPS logic function. Each display uses an “arm/fire” configuration requiring at least two deliberate operator actions. Operator use of the “arm” portion of the display triggers a plant alarm. The two manual opening schemes from the SSLC/ESF and from the DPS are diverse. Each safety VDU provides a display with an “arm/fire” switch (one per division) to manually initiate ADS as a system, rather than initiating each valve individually (IEEE Std 603, Sections 5.8, 6.2, and 7.2). If the operator uses any 2/4 “arm/fire” switches, the ADS sequence seals in and starts the ADS valve-opening sequence (IEEE Std 603, Section 5.2). This requires at least 2/4 deliberate operator actions.

DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the protection systems design is completed in compliance with IEEE Std 603. Based on the design implementing the IEEE Std 603 requirement for actuation seal-in provisions and requiring two deliberate operations to perform a manual operation, such as the reset function, their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the design acceptably addresses IE Bulletin 80-06.

7.1.1.3.6 Compliance with General Design Criteria Related to Instrumentation and Controls

SRP Table 7.1 and Appendix 7.1-A identify the following GDC as the acceptance criteria for I&C systems important to safety. This section provides a general evaluation of each GDC, while the remaining sections of Chapter 7 of this report evaluate the specific application of the GDC.

(1) GDC 1, “Quality standards and records”

GDC 1 requires quality standards and maintenance of appropriate records. The staff evaluated whether GDC 1 is adequately addressed for the DCIS in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states, for GDC 1, that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. The discussion of conformance to RGs and standards for 10 CFR 50.55a(a)(1) in Section 7.1.1.3.3 applies to GDC 1. Based on the finding for 10 CFR 50.55a(a)(1), the staff finds that the requirements of GDC 1 regarding RGs and standards are adequately addressed for the DCIS.

GDC 1 also includes requirements for a quality assurance program and the maintenance of appropriate records. Chapter 17 of this report addresses the evaluation of the applicant’s quality assurance program and appropriate records.

(2) GDC 2, “Design bases for protection against natural phenomena”

GDC 2 requires design bases for protection against natural phenomena. The staff evaluated whether GDC 2 is adequately addressed for the DCIS. SRP Table 7.1 and SRP Appendix 7.1-A show that GDC 2 applies to all I&C safety systems (DCD Sections 7.2, 7.3, 7.4, 7.5, and 7.6). SRP Appendix 7.1-A states, for GDC 2, that the design bases for protection against natural phenomena for I&C systems important to safety should be provided for the I&C system. DCD Tier 2, Revision 9, Table 7.1-1, identifies the applicability of GDC 2 to I&C systems, including its applicability to all safety I&C systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.5, states that the safety I&C systems are designed to meet the requirements of IEEE Std 603, Section 5.4, for EQ. DCD Tier 1, Revision 9, Section 3.8 includes the ITAAC for the applicant to confirm the EQ of safety electrical and digital I&C equipment, which is consistent with the requirements of IEEE Std 603, Section 5.4. Furthermore, the safety I&C systems are designed to meet the guidance in RG 1.100, Revision 3, September 2009, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” and IEEE Std 344, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations,” associated with seismic qualification.

SRP Appendix 7.1-A also states, for GDC 2, that the design bases should identify those systems and components that should be qualified to survive the effects of earthquakes and other natural phenomena. DCD Tier 2, Revision 9, Table 3.2-1, identifies that the safety I&C systems are designed as seismic Category I systems. DCD Tier 2, Revision 9, Chapter 3, states that the I&C systems important to safety are qualified for protection against natural phenomena, consistent with the analysis of these events, and that they are located and housed in structures consistent with these requirements. Section 3.10 of this report evaluates the adequacy of qualification programs to demonstrate the capability of I&C systems to withstand the effects of natural phenomena. DCD Tier 2, Revision 9, Section 3.11, specifies that instrumentation systems needed for severe accidents are designed to operate in the severe accident environment for which they are intended, and over the time span for which they are needed. Section 3.11 of this report evaluates DCD Tier 2, Revision 9, Section 3.11. DCD Tier 1, Revision 9, Section 3.8, includes the ITAAC to verify the EQ and seismic qualification of instrumentation systems needed for severe accidents. Based on the above, the staff finds that the requirements of GDC 2 are adequately addressed.

(3) GDC 4, “Environmental and dynamic effects design bases”

GDC 4 requires environmental and dynamic effects design bases. The staff evaluated whether GDC 4 is adequately addressed for the DCIS. SRP Table 7.1 and SRP Appendix 7.1-A show that GDC 4 applies to all I&C safety systems (DCD Sections 7.2, 7.3, 7.4, 7.5, and 7.6). SRP Appendix 7.1-A states, for GDC 4, that the environmental and dynamic effects (e.g., missiles) design bases for I&C systems important to safety should be provided for each system. DCD Tier 2, Chapter 7, discusses environmental design bases, but missile design bases are discussed in DCD Tier 2, Chapter 3, as described below. DCD Tier 2, Revision 9, Table 7.1-1, identifies the applicability of GDC 4 to I&C systems, including its applicability to all safety I&C systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.5, describes the methods used for temperature and humidity, radiation, and EMI qualification. Safety I&C systems are designed to meet the EQ requirements in RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants”, RG 1.100, and IEEE Std 323, “Qualifying Safety - Related Equipment for Nuclear Power Generating Stations”, associated with EQ. DCD Tier 1, Revision 9, Section 3.8, includes the ITAAC for the applicant to confirm the EQ of safety electrical and digital I&C equipment, which is consistent with the requirements of IEEE Std 603, Section 5.4.

SRP Appendix 7.1-A also states, for GDC 4, that the design bases should identify those systems and components that are qualified to accommodate the effects of environmental conditions and that are protected from the dynamic effects of missiles, pipe whipping, and discharging fluids. DCD Tier 2, Revision 9, Chapter 3, specifies that safety I&C systems be protected from the dynamic effects of missiles, pipe whipping, and discharging fluids. DCD Tier 2, Revision 9, Table 3.2-1, identifies the equipment classification of safety I&C systems. DCD Tier 2, Revision 9, Table 3.11-1, identifies the qualification program and required operating times for electrical and mechanical equipment, including safety I&C systems. DCD Tier 2, Revision 9, Appendix 3H, identifies the design bases environmental conditions by plant zone and typical equipment. Section 3.10 of this report evaluates the adequacy of qualification programs to demonstrate the capability of I&C systems to withstand dynamic effects.

Section 3.11 of this report evaluates the adequacy of EQ programs. DCD Tier 1, Revision 9, Section 3.8, provides the ITAAC for the EQ of the safety electrical equipment located in a harsh environment to verify that such equipment can perform its safety function under normal, abnormal, and DBA environmental conditions.

SRP Appendix 7.1-A also states, for GDC 4, that the I&C systems needed for severe accidents must be designed so there is reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Section 7.5.2.3, Item (4), of this report evaluates monitoring for severe accidents, which the staff finds is acceptable.

Based on the above, the staff finds that environmental and dynamic design bases are provided for the DCIS as a whole, that qualified systems are identified, and that the I&C systems needed for severe accidents are designed so there is reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Accordingly, the staff finds that the requirements of GDC 4 are adequately addressed.

(4) GDC 10, "Reactor design"

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the DCIS. SRP Table 7.1 identifies that GDC 10 applies to I&C protection and control systems (DCD Sections 7.2, 7.3, 7.6, and 7.7). SRP Appendix 7.1-A states, for GDC 10, that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. DCD Tier 2, Revision 9, Table 15.1-6, identifies systems, including protection, and control systems, required to mitigate AOOs. DCD Tier 2, Revision 9, Chapter 7, includes corresponding actions in the design bases of the protection and control systems to maintain the reactor core and reactor coolant systems within appropriate margins. In DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the protection and control system designs implement these design bases. Sections 7.2, 7.3, 7.6, and 7.7 of this report further discuss the implementation of GDC 10. Accordingly, because the applicant identifies necessary protection and safety actuations in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 10 are adequately addressed for the DCIS.

(5) GDC 13, "Instrumentation and control"

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the DCIS. SRP Table 7.1 identifies that GDC 13 applies to all I&C systems, including supporting systems (all DCD Chapter 7 sections). DCD Tier 2, Table 7.1-1, identifies that GDC 13 applies to all I&C systems important to safety. The applicant has identified interrelated processes to design the monitoring and control capabilities. NEDE-33226P and NEDE-33245P, as part of a software life cycle process, define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Section 3.2, includes the DAC/ITAAC for verifying that the software plans were developed and implemented consistent with this process and produce acceptable design outputs. DCD Tier 1, Section 3.3, includes the DAC/ITAAC for implementing an HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room.

In addition, the staff evaluated the appropriate controls to maintain these variables and systems within prescribed operating ranges for specific systems throughout Chapter 7 of this report and finds them acceptable. Accordingly, based on the defined processes for designing the monitoring and control capability, their verification in the DCD Tier 1, DAC/ITAAC, and the appropriate controls provided for specific systems, the staff finds that the requirements of GDC 13 are adequately addressed for the DCIS.

(6) GDC 15, "Reactor coolant system design"

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 15 applies to I&C protection and control systems (DCD Sections 7.2, 7.3, 7.6, and 7.7). SRP Appendix 7.1-A states, for GDC 15, that the staff review should evaluate the I&C system contributions to design margin for reactor coolant systems. DCD Tier 2, Table 15.1-6, identifies systems, including protection and control systems, required to mitigate AOOs. DCD Tier 2, Chapter 7, includes corresponding actions in the design bases of the protection and control systems to maintain the reactor core and reactor coolant systems within appropriate margins. DCD Tier 1, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the protection and control system designs implement these design bases. Sections 7.2, 7.3, 7.6, and 7.7 of this report further discuss the implementation of GDC 15. Accordingly, based on the applicant's identification of necessary protection and safety actuations in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 15 are adequately addressed for the DCIS.

(7) GDC 16, "Containment design"

GDC 16 requires containment leak-tight barrier against the uncontrolled release of radioactivity. The staff evaluated whether GDC 16 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 16 applies to I&C ESF and interlock logic (DCD Sections 7.3 and 7.6). SRP Appendix 7.1-A states that GDC 16 imposes functional requirements on ESF I&C systems to the extent that they support the requirement that the containment provide a leak tight barrier. SRP Appendix 7.1-A identifies several potential relevant I&C functions but the only one applicable to the ESBWR passive design is containment isolation. GDC 16 is not applicable to the ESBWR interlock logic, since the one interlock is associated with coolant injection into the reactor rather than containment isolation. DCD Tier 2, Revision 9, Section 7.3.2, describes the PCCS, which provides containment cooling. While the PCCS has no I&C functions, it does rely on I&C functions in the ICS to perform its safety functions, as described in DCD Tier 2, Revision 9, Section 7.4.4.3. DCD Tier 2, Revision 9, Sections 7.3.3 and 7.3.5 identify the containment isolation functions in the LD&IS and SSLC/ESF system design bases. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the ESF actuation and control systems, VBIF, and all subsystem designs implement these design bases. Accordingly, based on the applicant's identification of the necessary containment isolation functions in the design bases of the ESF actuation and control systems, VBIF, and all subsystems, and their verification in the DCD Tier 1 DAC/ITAAC, the NRC staff finds that the requirements of GDC 16 are adequately addressed for the DCIS.

(8) GDC 19, "Control room"

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 19 applies to all I&C systems and supporting systems (DCD Sections 7.2 through 7.8). The evaluation of the operation of specific I&C systems is included throughout Chapter 7 of this report. DCD Tier 2, Table 7.1-1, identifies that GDC 19 applies to all I&C systems, consistent with SRP Table 7-1. DCD Tier 2, Section 18.1.5, addresses the adequacy of the human factors aspects of the control room design and specifies that divisional separations for control, alarm, and display be maintained. For example, DCD Tier 2, Sections 6.4.8 and 9.4.1.5, detail CRHA design features, including instrumentation for air flow, differential pressure (across the area envelope), and safety radiation monitoring. Section 6.4 of this report evaluates control room habitability. DCD Tier 2, Section 7.4.2, describes the RSS capability. The RSS maintains Division I/II separation and isolation.

The I&C systems described in DCD Tier 2, Revision 9, Sections 7.2 through 7.8, include details of control interfaces through either the Q-DCIS or the N-DCIS. DCD Tier 2, Revision 9, Sections 7.1.3.1 and 7.1.4.2, identify the design bases for these control systems and their conformity to IEEE Std 603. The design includes full control functionality. As an example, the Q-DCIS supports safety system monitoring and operator input to and from the MCR and RSS. Protective actions associated with the RPS, neutron monitoring, and the SSLC/ESF can be manually initiated at the system level, in conformance with RG 1.62, and at the division level, in conformance with IEEE Std 603.

DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the control room functionality is implemented. Accordingly, based on a review of the design bases for control room functions, I&C interfaces, and DCD Tier 1 ITAAC for these functions, the staff finds that the requirements for GDC 19 are adequately addressed.

(9) GDC 20, "Protection system functions"

GDC 20 requires that the protection system be designed to (1) initiate automatically the operation of the appropriate systems, including reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs and (2) sense accident conditions and initiate the operation of systems and components important to safety. The staff evaluated whether GDC 20 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 20 applies to the protection systems (RTS and ESF) (DCD Sections 7.2 and 7.3). SRP Appendix 7.1-A states that GDC 20 is addressed for protection systems by conformance to IEEE Std 603, Sections 4, 5, 5.5, 6.1, 6.8, and 7.1. DCD Tier 2, Table 7.1-1, identifies that GDC 20 applies to the protection systems consistent with SRP Table 7-1. The applicant has committed to following the guidance of RG 1.105 and has provided a setpoint methodology in NEDE-33304P, which is evaluated in Section 7.1.4 of this report. The staff also evaluated, for the protection systems, design-basis requirements, general functional requirements, and system integrity, involving IEEE Std 603, Sections 4, 5, 5.5, 6.1, 6.8, and 7.1, in Sections 7.1.1.3.10, 7.2.3.1, and 7.3.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that setpoints of safety functions are defined, determined, and implemented, based on the defined setpoint methodology, and that the design is completed in compliance with IEEE Std 603. In response to RAI 14.3-265 S01, which was incorporated in DCD Revision 6, the applicant updated DCD Tier 1, Tables 2.2.15-1 and 2.2.15-2, to cover applicable requirements of IEEE Std 603. As explained in Section 7.1.1.3.10 of this report, DCD Tier 1, Section 2.2.15, design description identifies that some IEEE Std 603 criteria do not appear in Table 2.2.15-1 (and therefore do not appear in Table 2.2.15-2) because some IEEE Std 603 criteria do not require ITAAC consistent with NRC guidance or because the criteria are covered by other non-system ITAAC. Based on the applicant's identification of the necessary protection safety actuation in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 20 are adequately addressed for the DCIS.

(10) GDC 21, "Protection system reliability and testability"

GDC 21 requires that protection systems be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. The staff evaluated whether GDC 21 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 21 applies to the RTS and ESF systems and to the supporting systems (DCD Sections 7.2 and 7.3). SRP Appendix 7.1-A states that GDC 21 is addressed for protection systems by conformance to IEEE Std 603 criteria, except Sections 5.4, 6.1, and 7.1. DCD Tier 2, Table 7.1-1, identifies that the guidelines for periodic testing in RGs 1.22 and 1.118 apply to the protection systems. DCD Tier 2, Section 7.2.1, describes the conformance of the RPS to IEEE Std 603, which is evaluated in Section 7.2.3.1 of this report. DCD Tier 2, Section 7.3, describes the conformance of ESF actuation and control systems, VBIF, and all subsystems to IEEE Std 603, which is evaluated in Section 7.3.3.1 of this report. DCD Tier 1, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the protection systems' design is completed in compliance with IEEE Std 603. In particular, DCD Tier 1, Table 2.2.15-2, includes the DAC/ITAAC for verifying that IEEE Std 603, Sections 5.7 and 6.5 are met. DCD Tier 1, Section 3.2, includes the DAC/ITAAC to verify the implementation of the software development process. In response to RAI 14.3-265 S01, which was incorporated in DCD Revision 6, the applicant updated DCD Tier 1, Tables 2.2.15-1 and 2.2.15-2, to cover applicable requirements of IEEE Std 603. As explained in Section 7.1.1.3.10 of this report, DCD Tier 1, Revision 9, Section 2.2.15, design description identifies that some IEEE Std 603 criteria do not appear in Table 2.2.15-1 (and therefore do not appear in Table 2.2.15-2) because some IEEE Std 603 criteria do not require ITAAC consistent with NRC guidance or because the criteria are covered by other non-system ITAAC. Based on the applicant's identification of the necessary protection safety actuation in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 21 are adequately addressed for the DCIS.

(11) GDC 22, "Protection system independence"

GDC 22 requires, in the pertinent part, that protection systems be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function. The staff evaluated whether GDC 22 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 22 applies to the protection systems (RTS and ESF) and supporting systems (DCD Sections 7.2 and 7.3). SRP Appendix 7.1-A states that GDC 22 is addressed for protection systems by conformance to IEEE Std 603, Sections 4, 5.1, 5.3, 5.4, 5.5, 5.6, 6.2, 6.3, 6.8, 7.2, and 8. DCD Tier 2, Table 7.1-1, identifies that GDC 22 and RG 1.75 apply to the protection systems. DCD Tier 2, Section 7.2.1, describes the conformance of the RPS to IEEE Std 603, which is evaluated in Section 7.2.3.1 of this report. DCD Tier 2, Section 7.3, describes the conformance of ESF actuation and control systems, VBIF, and all subsystems to IEEE

Std 603, which is evaluated in Section 7.3.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the protection systems' design is completed in compliance with IEEE Std 603. In particular, DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that IEEE Std 603, Section 5.6, "Independence," is met. In response to RAI 14.3-265 S01, which was incorporated in DCD Revision 6, the applicant updated DCD Tier 1, Tables 2.2.15-1 and 2.2.15-2, to cover applicable requirements of IEEE Std 603. As explained in Section 7.1.1.3.10 of this report, DCD Tier 1, Revision 9, Section 2.2.15, design description identifies that some IEEE Std 603 criteria do not appear in Table 2.2.15-1 (and therefore do not appear in Table 2.2.15-2) because some IEEE Std 603 criteria do not require ITAAC consistent with NRC guidance or because the criteria are covered by other non-system ITAAC. Based on the applicant's identification of the necessary protection safety actuation in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 22 are adequately addressed for the DCIS.

(12) GDC 23, "Protection system failure modes"

GDC 23 requires that protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if certain conditions (such as disconnection of the system, loss of energy, or postulated adverse environments) are experienced. The staff evaluated whether GDC 23 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 23 applies to the protection systems (RTS and ESF) and supporting systems (DCD Sections 7.2 and 7.3). SRP Appendix 7.1-A states that GDC 23 is addressed for protection systems by conformance to IEEE Std 603, Section 5.5. DCD Tier 2, Table 7.1-1, identifies that GDC 23 applies to the protection systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.6, states that the RTIF-NMS platform fails to a tripped state. Hardware and software failures detected by self-diagnostics cause a trip signal to be generated in the RPS division in which the failure occurs. DCD Tier 2, Revision 9, Section 7.1.6.6.1.6, states that the SSLC/ESF fails to a state in which the activated component remains "as-is" to prevent a control-system-induced LOCA. For the same reason, hardware and software failures detected by self-diagnostics do not initiate a signal in a failed SSLC/ESF division. IEEE Std 603, Section 5.5, is implemented in DCD Tier 1, Revision 9, DAC/ITAAC through the EQ DAC/ITAAC in DCD Tier 1, Section 3.8, and the software development ITAAC in DCD Tier 1, Section 3.2. Accordingly, based on the conformance to the applicable guidance and IEEE Std 603 and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 23 are adequately addressed for the DCIS.

(13) GDC 24, "Separation of protection and control systems"

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 24 applies to all I&C systems (DCD Sections 7.2 through 7.8). SRP Appendix 7.1-A states that GDC 24 is addressed for protection systems by conformance to IEEE Std 603, Sections 5.1, 5.6, 5.12, 6.3, 6.6, and 8, and particularly Sections 5.6 and 6.3. SRP Section 7.7 states that, for control systems isolated from safety systems, the applicable IEEE Std 603 sections are Sections 5.6.3 and 6.3.

DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 24 applies to all I&C systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.2 states, for IEEE Std 603, Section 5.1, that communication between safety control systems and nonsafety control systems is electrically isolated and one-way (which references DCD Tier 2, Revision 9, Section 7.1.3.3.6). DCD Tier 2, Revision 9, Section 7.1.3.3, states that safety fiber optic CIMs provide the safety isolation and separation and are qualified safety components. In RAI 7.1-65, the staff asked the applicant to describe the CIM safety-related functions and how they will be confirmed. RAI 7.1-65 was being tracked as an open item in the SER with open items. In its responses, the applicant clarified that CIMs are safety signal isolation devices, and the applicant revised DCD Tier 1, Table 2.2.15-2, Item 10, to provide DAC/ITAAC to verify that the software project's interdivisional communication systems have optically isolated fiber optical communication pathways. The staff finds that the responses are acceptable since the applicant clarified the CIM safety-related functions and how they will be confirmed. Based on the applicant's response, RAI 7.1-65 is resolved. In RAI 7.1-132, the staff requested that the applicant clarify which system contains the trip circuit to cut off power to the N-DCIS and to add this safety trip to the appropriate section of DCD Tier 1. RAI 7.1-132 was being tracked as an open item in the SER with open items. In its response, the applicant clarified the function of the safety Control Room Habitability Area HVAC Subsystem (CRHAVS) emergency trip circuit for the N-DCIS equipment. The applicant updated DCD Tier 1, Tables 2.2.13-2 and 2.2.13-3, and DCD Tier 2, Sections 6.4.8, 7.3.4.2, and 9.4.1.5, and incorporated these sections into in DCD Revision 6. The staff finds that the responses are acceptable since the applicant clarified in the DCD the system that contains the trip circuit to cut off power to predefined N-DCIS and non DCIS loads. Based on the applicant's response, RAI 7.1-132 is resolved. DCD Tier 2, Section 7.1.6.6.1, describes conformance to IEEE Std 603, Sections 5.1, 5.12, 6.6, 8.1, 8.2, and 8.3, respectively. In response to RAI 14.3-265 S01, which was incorporated in DCD, Revision 6, the applicant updated DCD Tier 1, Tables 2.2.15-1 and 2.2.15-2, to cover applicable requirements of IEEE Std 603. As explained in Section 7.1.1.3.10 of this report, DCD Tier 1, Revision 9, Section 2.2.15, design description identifies that some IEEE Std 603 criteria do not appear in Table 2.2.15-1 (and therefore do not appear in Table 2.2.15-2) because some IEEE Std 603 criteria do not require ITAAC consistent with NRC guidance or because the criteria are covered by other non-system ITAAC. Based on the applicant's identification of the necessary protection safety actuation in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 24 are adequately addressed for the DCIS.

(14) GDC 25, "Protection system requirements for reactivity control malfunctions"

GDC 25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. The staff evaluated whether GDC 25 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 25 applies to the RPS and control system interlock logic (DCD Sections 7.2 and 7.6). SRP Appendix 7.1-A states that GDC 25 is addressed for protection systems by conformance to IEEE Std 603, Section 4, which is associated with safety system design-basis requirements. DCD Tier 2, Revision 9, Section 7.2.1, provides the design bases for the RPS, which include protection from AOOs, such as continuous control rod withdrawal. DCD Tier 2, Revision 9, Chapter 15, includes an analysis for continuous rod withdrawal in several scenarios to show that the RPS is designed to prevent fuel design limits from being exceeded. DCD Tier 1, Revision 9, Tables 2.2.1-6 and 2.2.7-4, provide the ITAAC for verification of these reactor protection functions. In response to RAI 14.3-265 S01, which was incorporated in DCD Revision 6, the applicant updated DCD Tier 1, Tables 2.2.15-1 and 2.2.15-2, to cover the applicable requirements of IEEE Std 603. As explained in Section 7.1.1.3.10 of this report, DCD Tier 1, Section 2.2.15, design description identifies that the ITAAC do not include IEEE Std 603, Criteria 4.2, 4.3, 4.10, 4.11, and 4.12, because SRP Section 14.3.5, and RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Section C.II.1, do not include these criteria as ITAAC. DCD Tier 2, Revision 9, Section 7.1.6.6.1.1, describes how IEEE Std 603, Criteria 4.2, 4.3, 4.10, 4.11, and 4.12 are addressed in the DCD. The staff finds the explanation of IEEE Std 603, Criteria 4.2, 4.3, 4.10, 4.11, and 4.12 and their exclusion from the ITAAC acceptable. Based on the applicant's identification of the necessary protection safety actuation in the design bases for the protection and control systems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 25 are adequately addressed for the DCIS.

(15) GDC 28, "Reactivity limits"

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. The staff evaluated whether GDC 28 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 28 applies to I&C interlock logic and control systems (DCD Sections 7.6 and 7.7). SRP Appendix 7.1-A states that GDC 28 imposes functional requirements on I&C interlock and control systems to the extent that they are provided to limit reactivity increases to prevent or limit the effect of reactivity accidents. DCD Tier 2, Revision 9, Sections 7.7.2 and 7.7.6 state that the RC&IS and NMS conform to GDC 28. DCD Tier 2, Sections 7.7.2 and 7.7.6, and DCD Tier 2, Revision 9, Chapter 15, Tables 15.1-5 and 15.1-6 identify RC&IS actuations and other actions that reduce the need for the actuation of protection and safety systems to mitigate AOOs. DCD Tier 2, Section 7.7.2, includes corresponding actions in the design bases of the RC&IS to provide appropriate limits on the potential amount and rate of increase in reactivity. In particular, DCD Tier 2, Section 7.7.2.2.7.4, identifies the control rod block functions performed by the RC&IS. DCD Tier 2, Section 7.7.6, describes the MRBM, which monitors more than one region of the core and provides input to the RC&IS. DCD Tier 1, Revision 9, Section 2.2.1, includes the ITAAC for the applicant to verify that the as-built RC&IS implements these actions. Accordingly, based on identified RC&IS actions and their verification in the ITAAC, the staff finds that the requirements of GDC 28 are adequately addressed for these nonsafety systems.

(16) GDC 29, "Protection against anticipated operational occurrences"

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 29 applies to the protection systems, control systems, and supporting systems (DCD Sections 7.2, 7.3, and 7.7). SRP Appendix 7.1-A states that GDC 29 is addressed by conformance, as applicable, to GDC 20–25 and GDC 28. In DCD Tier 2, Revision 9, Table 7.1-1 and Sections 7.2 and 7.7 identify that GDC 29 applies to the applicable RPS and control systems. However, DCD Tier 2, Table 7.1-1 and Section 7.3 do not identify that GDC 29 applies to the applicable ESF actuation and control systems, VBIF, and all subsystems. In RAI 7.3-14, the staff asked the applicant to clarify the applicability of GDC 29. RAI 7.3-14 was being tracked as an open item in the SER with open items. In response to RAI 7.3-14, the applicant stated that it would address conformance to GDC 29 in the response to RAI 7.1-99. The applicant incorporated into DCD Revision 6 the responses to RAI 14.3-265, RAI 14.3-265 S01, RAI 7.1-99, 7.1-100, and 7.1-101, which correct the inconsistencies in DCD Tier 1 and Tier 2. The applicant updated Table 7.1-1 to cover all applicable criteria including GDC for all safety-related systems. The staff finds that the response to RAI 7.3-14 was acceptable, as augmented by the response to RAI 7.1-99, since the applicant clarified the applicability of GDC 29. Based on the applicant's responses, RAI 7.3-14 is resolved. Based on the applicant's identification of the necessary protection safety actuation in the design bases for the protection and control systems, the verification of the as-built system by the DAC/ITAAC in DCD Tier 1, Revision 9, and verification that the DCIS complies with GDC 20-25 and 28 as discussed above, the staff finds that the requirements of GDC 29 are adequately addressed for the DCIS.

(17) GDC 33, "Reactor coolant makeup"

GDC 33 requires a system to supply reactor coolant makeup for protection against small breaks in the RCPB. SRP Appendix 7.1-A states that GDC 33 imposes functional requirements on the ESF I&C systems provided to initiate, control, and protect the integrity of reactor coolant makeup systems to protect against small breaks in the RCPB. GDC 33 also requires that necessary I&C systems be operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Section 3.1.4.4, states that GDC 33 applies to the CRD hydraulic system, described in DCD Tier 2, Section 4.6.1.2.4; the ICS, described in DCD Tier 2, Section 5.4.6; and the ADS and GDCS, described in DCD Tier 2, Section 6.3. The CRD hydraulic system is nonsafety. DCD Tier 2, Sections 7.4.4 and 7.3.1, identify the corresponding reactor coolant makeup initiation, control, and protection functions in the design bases. The performance and reliability requirements of GDC 33 are addressed by the applicability of IEEE Std 603 to the ICS, ADS, and GDCS. In particular, Sections 5.1, 5.7, and 6.5 of IEEE Std 603 provide requirements for single failures and testability. Sections 7.3 and 7.4 of this report evaluate conformance of these systems to IEEE Std 603. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the ICS, ADS, and GDCS designs implement these design bases and conform to IEEE Std 603. DCD Tier 2, Revision 9, Section 8.1.3, states that the Q-DCIS, which includes the ICS, ADS, and GDCS I&C systems, is powered normally by the safety isolation power centers, or by the safety 250 volt dc batteries for 72 hours, if normal power is lost. Therefore, these systems are operable using either onsite or offsite power (assuming that only one source is available). Chapter 8 of this report evaluates the safety 250 volt dc power distribution system, including batteries. Accordingly, based on the applicant's identification of necessary reactor coolant makeup functions in the design bases of the ICS, ADS, and GDCS and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 33 are adequately addressed for the DCIS.

(18) GDC 34, "Residual heat removal"

GDC 34 requires a system to remove residual heat. SRP Appendix 7.1-A states that GDC 34 imposes functional requirements on the ESF, safe-shutdown, and interlock I&C systems provided to initiate, control, and protect the integrity of residual heat removal systems. GDC 34 also requires that the necessary I&C systems be operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Section 3.1.4.5, states that GDC 34 applies to the ICS, described in DCD Tier 2, Section 5.4.6. DCD Tier 2, Section 7.4.4, identifies the corresponding residual heat removal initiation, control, and protection functions in the design bases. The performance and reliability requirements of GDC 34 are addressed by the application of IEEE Std 603 to the ICS. In particular, Sections 5.1, 5.7, and 6.5 of IEEE Std 603 provide requirements for single failures and testability. Section 7.4 of this report evaluates conformance of the ICS to IEEE Std 603. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the ICS design implements these design bases and conforms to IEEE Std 603. DCD Tier 2, Revision 9, Section 8.1.3, states that the Q-DCIS, which includes the ICS I&C system, is normally powered by the safety isolation power centers, or, if normal power is lost, by the safety 250 volt dc batteries for 72 hours. Therefore, these systems are operable using either onsite or offsite power (assuming that only one source is available). Chapter 8 of this report evaluates the safety 250 volt dc power distribution system, including batteries. Accordingly, based on the applicant's identification of the necessary residual heat removal functions in the design bases of the ICS and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 34 are adequately addressed for the DCIS.

(19) GDC 35, "Emergency core cooling"

GDC 35 requires a system to provide abundant emergency core cooling. SRP Appendix 7.1-A states that GDC 35 imposes functional requirements on the ESF, safe-shutdown, and interlock I&C systems provided to initiate, control, and protect the integrity of ECCS. GDC 35 also requires that the necessary I&C systems be operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Section 3.1.4.6, states that GDC 35 applies to the ECCS, including the ICS, SLC system, GDCS, and ADS, as described in DCD Tier 2, Section 6.3. DCD Tier 2, Sections 7.4.4 (ICS), 7.4.1 (SLC System), and 7.3.1 (ADS and GDCS) identify the corresponding ECCS initiation, control, and protection functions in the design bases. The performance and reliability requirements of GDC 35 are addressed by the application of IEEE Std 603 to the ECCS. In particular, Sections 5.1, 5.7, and 6.5 in IEEE Std 603 provide requirements for single failures and testability. Sections 7.4 and 7.3 of this report evaluate conformance of the ECCS to IEEE Std 603. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the ECCS designs implement these design bases and conform to IEEE Std 603. DCD Tier 2, Revision 9, Section 8.1.3, states that the Q-DCIS, which includes the ICS, SLC system, GDCS, and ADS I&C systems, is normally powered by the safety isolation power centers, or, if power is lost, by safety 250 volt dc batteries for 72 hours. Therefore, these systems are operable using either onsite or offsite power (assuming that only one source is available). Chapter 8 of this report evaluates the safety 250 volt dc power distribution system, including batteries. Accordingly, based on the applicant's identification of the necessary ECCS functions in the design bases of the ICS, SLC system, ADS, and GDCS and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 35 are adequately addressed for the DCIS.

(20) GDC 38, "Containment heat removal"

GDC 38 requires a system to remove heat from the reactor containment. SRP Appendix 7.1-A states that GDC 38 imposes functional requirements on the ESF, safe-shutdown, and interlock I&C systems provided to initiate, control, and protect the integrity of containment heat removal systems. GDC 38 also requires that the necessary I&C systems be operable using either onsite or offsite power (assuming only one source is available). DCD Tier 2, Section 3.1.4.9, states that GDC 38 applies to the PCCS, described in DCD Tier 2, Section 6.2.2. In DCD Tier 2, Sections 6.2.2 and 7.3.2 state that the PCCS does not have instrumentation, control logic, or power-actuated valves and does not need or use electrical power for its operation in the first 72 hours after a LOCA. While the PCCS has no I&C functions, it does rely on I&C functions in other systems, namely the ICS, SSLC/ESF, and FAPCS, to perform its safety functions. In RAI 7.1-140, the staff requested the applicant to clarify the active components, electrical motive power, and I&C functions needed for the PCCS to perform its safety functions, including supporting functions provided by other systems. RAI 7.1-140 was being tracked as an open item in the SER with open items. In its response, the applicant clarified that the PCCS relies on the water in the equipment storage pool and reactor well to perform its safety functions for

72 hours. Pool cross-connect valves are active components that open to allow water in the equipment storage pool and reactor well to flow into the IC/PCCS pools. FAPCS provides four safety-related level sensors in each IC/PCCS inner expansion pool. The cross-connect valves are opened when the sensors detect a low level condition in either pool. The FAPCS also provides four nonsafety level sensors in each inner expansion pool which are used by DPS to open the cross-connect valves. The air-operated cross-connect valves require pneumatic and electrical motive power to open, which is provided by a pneumatic accumulator, and the safety-related UPS. The squib cross-connect valves are opened pyrotechnically and need only electrical motive power to open, which is provided by the safety-related UPS. The response modified multiple sections of DCD Tier 2, including Sections 7.4.4.3, 7.5.5, and 7.8.1.2.5, to address these clarifications. The staff finds the response is acceptable since the applicant identified the necessary support systems and function for the PCCS to perform its safety functions. Based on the applicant's response, RAI 7.1-140 is resolved.

As noted above, the response to RAI 7.1-140 added containment heat removal initiation, control, and protection functions to the design bases in DCD Tier 2, Section 7.4.4.3. The performance and reliability requirements of GDC 38 are addressed by the application of IEEE Std 603 to the ICS. In particular, Sections 5.1, 5.7, and 6.5 of IEEE Std 603 provide requirements for single failures and testability. Section 7.4 of this report evaluates conformance of the ICS to IEEE Std 603. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the ICS design implements these design bases and conforms to IEEE Std 603. DCD Tier 2, Revision 9, Section 8.1.3, states that the Q-DCIS, which includes the ICS I&C system, is normally powered by the safety isolation power centers, or, if normal power is lost, by the safety 250 volt dc batteries for 72 hours. Therefore, these systems are operable using either onsite or offsite power (assuming that only one source is available). Chapter 8 of this report evaluates the safety 250 volt dc power distribution system, including batteries. Accordingly, based on the applicant's identification of the necessary containment heat removal functions in the design bases of the ICS, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 38 are adequately addressed for the DCIS.

(21) GDC 41, "Containment atmosphere cleanup"

GDC 41 requires systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment. SRP Appendix 7.1-A states that GDC 41 imposes functional requirements on the ESF and interlock I&C systems provided to initiate, control, and protect the integrity of the containment atmosphere cleanup systems. GDC 41 also requires that the necessary I&C systems be operable using either onsite or offsite power (assuming that only one source is available). The staff evaluated whether GDC 41 is adequately addressed for the DCIS. SRP Table 7-1 identifies that GDC 41 applies to the ESF and interlock I&C systems (DCD Sections 7.3 and 7.6). DCD Tier 2, Revision 9, Section 6.5.4, states that the suppression pool performs a fission product cleanup function in conformance with GDC 41. However, this function does not require any I&C functions. DCD Tier 2, Revision 9, Section 6.2.5.1, states that safety combustible gas control is provided by an inerted containment; therefore, GDC 41 is not applicable to the design for this function. Section 6.5 of this report evaluates the containment atmosphere cleanup systems with regard to GDC 41. Accordingly, because the atmospheric cleanup systems do not require any I&C functions, the staff finds that the requirements of GDC 41 do not apply to the I&C design.

(22) GDC 44, "Cooling water"

GDC 44 requires a system to transfer heat from structures, systems, and components (SSCs) important to safety, to an ultimate heat sink. According to SRP Appendix 7.1-A, GDC 44 imposes functional requirements on the ESF, interlock, and control I&C systems provided to initiate, control, and protect the integrity of cooling water systems important to safety that transfer heat to the ultimate heat sink. GDC 44 also requires that necessary I&C systems be operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Sections 3.1.4.15 and 9.2.5, state that the IC/PCCS pools are the ultimate heat sink. These sections also state that the IC/PCCS pools have no active components and do not require I&C functions to perform their safety function of transferring heat to the atmosphere. Accordingly, because the ultimate heat sink cooling water does not require any I&C functions, the staff finds that the requirements of GDC 44 do not apply to the I&C design.

7.1.1.3.7 Staff Requirement Memorandum Issues

SECY-93-087 identified two digital I&C-related issues designated as SRM issues:

(1) SRM to SECY-93-087, Item II.Q, "Defense Against Common - Mode Failures in Digital Instrumentation and Control Systems."

(2) SRM to SECY-93-087, Item II.T, "Control Room Annunciator (Alarm) Reliability."

The staff evaluated whether the guidelines of SRM to SECY-93-087, Item II.Q, are adequately addressed for the DCIS. SRP Table 7-1 identifies that the Item II.Q applies to the protection systems, ESF actuation systems, control systems, and the DPS (DCD Sections 7.2, 7.3, 7.7, and 7.8).

DCD Tier 2, Revision 9, Table 7.1-1, identifies that the SRM to SECY-93-087, Item II.Q, applies to the applicable systems. NEDO-33251 provides the primary assessment of conformance to the guidelines Item II.Q, along with BTP HICB-19. Section 7.1.3 of this report documents the staff's positions on Item II.Q and the staff's evaluation of the D3 assessment.

The staff evaluated whether the guidelines of SRM to SECY-93-087, Item II.T, are adequately addressed for the DCIS. SRP Table 7-1 identifies that Item II.T applies to information systems important to safety and supporting systems (DCD Section 7.5).

The staff position in Item II.T is as follows: (a) The annunciator system is considered to consist of sets of alarms (which may be displayed on tiles, VDUs, or other devices) and sound equipment; logic and processing support; and functions to enable operators to silence, acknowledge, reset, and test alarms. (b) The MCR should contain compact, redundant operator workstations with multiple display and control devices that provide organized, hierarchical access to alarms, displays, and controls. Each workstation should have the full capability to perform MCR functions as well as support division of tasks between two operators. (c) The display and control features should be designed to satisfy existing regulations, for example, separation and independence requirements for Class 1E circuits (IEEE Std 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits"), and specifications for manual initiation of protective actions at the systems level per RG 1.62. The designer should use existing defensive measures (e.g., segmentation, fault tolerance, signal validation, self-testing, error checking, supervisory watchdog programs), as appropriate, to assure that alarm, display, and control functions provided by the redundant workstations meet these criteria. (d)

Alarms that are provided for manually controlled actions for which no automatic control is provided, and that are required for the safety systems to accomplish their safety functions, should meet the applicable specifications for Class 1E equipment and circuits.

DCD Tier 2, Revision 9, Table 7.1-1, identifies that the SRM to SECY-93-087, Item II.T, applies to applicable information systems important to safety, the Q-DCIS, and the N-DCIS. DCD Tier 2, Revision 9, Section 7.1.6.3, states that the AMS follows guidance in Item II.T, for redundancy, independence, and separation, because the "alarm system" is considered redundant (i.e., includes redundant features). Alarm points are sent through dual networks to redundant message processors on dual power supplies. The processors are dedicated only to performing alarm processing. The alarms are displayed on multiple independent VDUs that each have dual power supplies. The alarm tiles, or their equivalent, are driven by redundant datalinks (with dual power). There are redundant alarm processors. No alarms require manually controlled actions for safety systems to accomplish their function. Thus, the requirements for safety equipment and circuits do not apply.

SRP Appendix 7.1-A states that alarms that are provided for manually controlled action for which no automatic control is provided, and that are required for the safety systems to accomplish their safety function, should meet the applicable specifications for Class 1E equipment and circuits. Because the design is a passive plant, all of the safety systems are initiated automatically. No preplanned manual controlled action is required for the safety systems. The staff finds that the exception documented in DCD Tier 2, Revision 9, Section 7.1.6.3 and Table 1.9-7 (alarm is not classified as Class 1E equipment or circuits), is acceptable. Section 7.5 of this report provides a further evaluation of alarm (annunciator) systems. In addition, in DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, 3.7, and 3.8, include the DAC/ITAAC for the applicant to verify that the I&C systems design conforms to the applicable regulations. Based on the above, the staff finds that the guidelines of the SRM to SECY-93-087, Item II.T, are adequately addressed for the DCIS.

7.1.1.3.8 Regulatory Guides

In DCD Tier 2, Revision 9, Table 7.1-1, the applicant identified the applicable regulatory requirements, including the applicable RGs, for each of the DCIS systems. DCD Tier 2, Table 7.1-1, identifies that the RGs apply to the applicable systems identified in SRP Table 7.1 with the exception of RGs 1.174, 1.177, 1.189, and 1.200. Section 7.1.1.3.2 of this report provides a general discussion of differences in applicability between DCD Tier 2, Table 7.1-1 and SRP Table 7.1.

SRP Table 7.1 does not identify any applicability for RGs 1.174, 1.177, and 1.200, but refers to BTP HICB-12, "Guidance on Establishing and Maintaining Instrument Setpoints," instead. BTP HICB-12 identifies that RGs 1.174, 1.177, and 1.200 provide guidance for using a risk-informed approach to evaluate changes to instrument calibration and surveillance test intervals for reasons other than a 24-month fuel cycle. DCD Tier 2, Revision 9, Table 1.9-21, identifies that these RGs are not applicable to the design certification but indicates that they may be used by Licensees. Accordingly, the applicant does not apply these approaches to the design for calibration intervals, which the staff finds acceptable.

SRP Table 7.1 identifies that RG 1.189 is applicable to SAR Chapter 7.4. SRP Section 7.4 identifies RG 1.189 as a reference and discusses RG 1.189 in a footnote concerning remote shutdown capability and the assumptions to be used in the case that smoke from a fire requires the evacuation of the MCR. The footnote also states that conformance to RG 1.189 is evaluated with SRP Section 9.5.1. DCD Tier 2, Revision 9, Section 9.5.1, describes conformance of the fire protection program to RG 1.189, which is evaluated in Section 9.5.1 of this report. Accordingly, the staff finds the above approach acceptable.

DCD Tier 2, Table 1.9-21b, identifies no exceptions to I&C-related RGs. DCD Tier 2, Revision 9, Table 1.9-7, identifies differences with the SRPs with regard to RGs 1.22, 1.118, and 1.151.

DCD Tier 2, Table 1.9-7, identifies that some actuators and digital sensors, because of their locations, cannot be fully tested during actual reactor operation, in conformance with RG 1.22. DCD Tier 2, Revision 9, Section 7.1.6.4, states that such equipment is identified and provisions for meeting the guidance of Paragraph D.4 (in accordance with BTP HICB-8) are discussed within the safety evaluation sections of DCD Tier 2, Revision 9, Sections 7.2 through 7.8. In DCD Tier 2, Sections 7.2.1.3.4, 7.3.1.1.3.4, 7.3.1.2.3.4, and 7.3.6.3.4, identify alternatives to full system testing during reactor operations for the RPS, ADS, GDCS, and VBIF system, respectively.

In RAI 7.3-17, the staff requested the applicant to describe how the VBIF conforms to RG 1.22 and BTP HICB-8. The staff raised this question since DCD Section 7.3.6.4 then stated that VB isolation function equipment inside containment is tested during refueling outages, but did not state why it is not practicable to test during reactor operation. RG 1.22 provides guidelines for justifying not testing actuated equipment during reactor operations. RAI 7.3-17 was being tracked as an open item in the SER with open items. In its response, the applicant modified the DCD to specify VBIF testing during reactor operation consistent with RG 1.22, which addresses BTP HICB-8, “Guidance for Application of Regulatory Guide 1.22.” The staff finds the response is acceptable since the applicant clarified conformance to RG 1.22. Based on the applicant’s response, RAI 7.3-17 is resolved.

Consistent with RG 1.22, Regulatory Position D.4, the applicant has shown for the RPS, ADS, and GDCS, that no practicable system design would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant. For example, testing during operation would result in reactor scram or releases from the RCPB. The alternative testing includes (1) testing the RPS in overlapping stages, (2) testing the ADS and GDCS components during outages, and (3) testing the ADS and GDCS (described in DCD Tier 2, Revision 9, Section 6.3.2.7.4) squib initiators in a laboratory after removal from the squib valves. The staff finds these alternative testing approaches consistent with RG 1.22, Regulatory Position D.4 and therefore acceptable.

DCD Tier 2, Revision 9, Table 1.9-7, identifies clarifications and testing exceptions to RG 1.118. DCD Tier 2, Revision 9, Section 7.3.1.1.3.4, identifies that a full functional test of the ADS is not practical because a LOCA results if the non re-closable DPVs are opened. Acceptable reliability of equipment operation is demonstrated by alternate test methods. System logic is periodically self-tested, and initiating circuits are continuously monitored. DPV valve initiators periodically are removed and test-fired in a laboratory. RPV level transmitters are located outside containment so calibration verification can be performed during plant operation. The staff finds this acceptable.

DCD Tier 2, Revision 9, Table 1.9-7, identifies that RG 1.151 does not apply to the SB&PC system and the N-DCIS. DCD Tier 2, Revision 9, Section 7.1.5.3.4, identifies that the N-DCIS receives signals from sensors in various systems in the plant, which are from instrument sensing lines from nonsafety instrumentation, but the N-DCIS itself does not contain instrument sensing lines. DCD Tier 2, Revision 9, Section 7.7.5.3.3, identifies that the SB&PC system receives pressure signals from sensors in the NBS and the Main Condenser and Auxiliaries System but does not itself contain instrument sensing lines. The staff finds this acceptable.

7.1.1.3.9 Branch Technical Positions

In DCD Tier 2, Revision 9, Table 7.1-1, the applicant identifies the applicable regulatory requirements and guidance, including the applicable SRP BTPs, for each of the DCIS systems. The staff compared DCD Tier 2, Table 7.1-1, to SRP Table 7.1, and finds that the applicant has either documented the applicability of the guidance or addressed any exceptions, as discussed in Section 7.1.1.3.3 of this report.

In DCD Tier 2, Revision 9, Table 1.9-7, the applicant states that the approach to software management and quality assurance complies with the intent of the SRP and BTP HICB-14, but is implemented in a set of acceptable equivalent alternative and mutually consistent plans which, applied in total, comprise the general requirements. Section 7.1.2 of this report addresses the staff's review of software development activities. DCD Tier 1, Revision 9, Section 3.2, documents the DAC/ITAAC for the software development process. DCD Tier 2, Revision 9, Section 7.1.6.5, generally discusses the conformance of the design to the BTPs listed in DCD Tier 2, Revision 9, Table 7.1-1. This report discusses conformance to BTPs throughout Chapter 7. The staff finds that the applicant has adequately addressed conformance with the listed SRP BTPs for the DCIS.

7.1.1.3.10 IEEE Standard 603 Requirements

The staff evaluated whether the applicant has adequately addressed all of the criteria listed in IEEE Std 603, as required by 10 CFR 50.55a(h)(3). As discussed in Section 7.1.1.3.1 of this report, the applicant is using the DAC approach to comply with IEEE Std 603 requirements. To implement this approach, the staff evaluated whether the applicant specified conformance to IEEE Std 603, consistent with the acceptance criteria in SRP Appendix 7.1-C. The staff also evaluated whether DCD Tier 1 included sufficient DAC to confirm that the completed design meets IEEE Std 603 requirements.

The staff also evaluated, in parallel, whether additional applicable guidance in IEEE Std 7-4.3.2 for safety systems using digital programmable computers is addressed. The staff used the acceptance criteria in SRP Appendix 7.1-D for criteria related to IEEE Std 7-4.3.2. However, in RAI 7.1-99, Item D, the staff requested that the IEEE Std 7-4.3.2 criteria not already covered by the DAC/ITAAC in DCD Tier 1, Sections 2.2.15 or 3.2, be included in that DAC/ITAAC. RAI 7.1-99 was being tracked as an open item in the SER with open items. In its response to RAI 14.3-265 S01, and RAIs 7.1-99, 7.1-100, and 7.1-101, all of which were incorporated in DCD Revision 6, the applicant corrected the inconsistent documentation in DCD Tier 1 and Tier 2. The applicant updated DCD Tier 1, Table 2.2.15-1, to include applicable IEEE Std 603 criteria for all safety I&C systems. Table 2.2.15-2 identifies design commitment to each IEEE Std 603 criterion for the software projects. As explained in Section 7.1.1.3.10 of this report, DCD Tier 1, Revision 9, Section 2.2.15, design description identifies that some IEEE Std 603 criteria do not appear in Table 2.2.15-1 (and therefore do not appear in Table 2.2.15-2) because some IEEE Std 603 criteria do not require ITAAC consistent with NRC guidance or because the criteria are covered by other non-system ITAAC. The ITAAC acceptance criteria contain two phases: (1) the DAC phase, which specifies the software projects design requirements, and (2) the ITAAC implementation phase, which specifies the methods to verify that the as-built design has satisfied the IEEE Std 603 requirements. In addition, the applicant significantly augmented the discussion of compliance with IEEE Std 603 sections in DCD Tier 2, Section 7.1.6.6.1; revised

DCD Tier 2, Table 7.1.1 and 7.1.2 to more clearly show conformance to regulatory requirements and IEEE Std 603; and implemented a consistent designation of systems and their conformance to requirements and guidelines throughout Chapter 7. Based on the review of DCD Tier 1, Revision 9, Sections 2.2.15 and 3.2, and the information referenced by DCD Tier 2, Revision 9, Table 7.1-2, the staff finds that the DCD has properly addressed compliance with IEEE Std 7-4.3.2. Based on the above and the applicant's response, RAIs 14.3-265, 7.1-99, 7.1-100, and 7.1-101 are resolved.

DCD Tier 2, Revision 9, Table 7.1-2, identifies specific sections of DCD Chapters 7 in which compliance with IEEE Std 603 is discussed.

7.1.1.3.10.1 *IEEE Standard 603, Section 4, "Safety System Designation"*

The staff evaluated whether IEEE Std 603, Section 4, is adequately addressed using SRP Appendix 7.1-C, Section 4. IEEE Std 603 states that, "The [safety system] design basis shall also be available as needed to facilitate the determination of the adequacy of the safety system design." SRP Appendix 7.1-C, Section 4, identifies characteristics that the safety system design basis should exhibit: completeness, consistency, correctness, traceability, unambiguity, and verifiability. For the completeness characteristic, SRP Appendix 7.1-C states, "As a minimum each of the design basis aspects identified in IEEE Std 603, Clauses [i.e. Sections] 4.1 through 4.12 should be addressed." DCD Tier 2, Revision 9, Section 7.1.6.6.1.1, provides a general discussion of conformance to IEEE Std 603, Section 4, and identifies several sections where high level safety system functional information is provided. DCD Tier 2, Revision 9, Table 1.3-1, defines the reactor system design characteristics. Tables 15.0-3, 15.0-4, 15.0-5, and 15.0-6 define the safety analysis acceptance criteria for the AOOs, infrequent events, special events, and accidents. Table 15.1-2 defines the operating modes for the entire operating envelope. Table 15.1-3 defines the abnormal events with applicable operating modes. Table 15.2-1 defines the input parameters, initial conditions and bounding limits for ATWS events and infrequent events. Table 15.5-2 defines the initial conditions, and bounding limits for ATWS events. DCD Tier 2, Revision 9, Sections 15.2, 15.3, 15.4 and 15.5 describe credited systems, interlocks, and functions for each DBE. Various sections of DCD Tier 2, Revision 9, Chapter 7, particularly Sections 7.2 through 7.5, document safety system design-basis descriptions.

The staff evaluated whether IEEE Std 603, Section 4.1, is adequately addressed for the safety systems. This criterion requires the identification of the DBEs applicable to each mode of operation, along with the initial conditions and allowable limits of plant conditions for each such event. The information to be provided should be consistent with the analysis in DCD Tier 2, Chapter 15. The analysis should include how the Q-DCIS automatically initiates appropriate protective action when a condition monitored by the system reaches a preset level. DCD Tier 2, Revision 9, Section 7.1.6.6.1.1, provides a general discussion of conformance to Section 4 and identifies several sections that provide high level safety system functional information. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 4.1 to verify that DBE information is incorporated into software projects during the software life cycle process. Based on the review of DCD Tier 2, Revision 9, the Chapter 15 tables listed above, and the review of DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.1 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.2, is adequately addressed for the safety systems. This criterion requires the identification of the safety functions and corresponding protective actions of the execute features for each DBE as part of the design basis. DCD Tier 2, Revision 9, Section 7.1.6.6.1.1, provides a general discussion of conformance to IEEE Std 603, Section 4, and identifies several sections that provide high level safety system functional information. DCD Tier 2, Revision 9, Table 15.1-6, defines the automatic safety instrument trips in response to each event. Safety design bases for each system are discussed in DCD Tier 2, Revision 9, Chapter 7, within the safety evaluation section for each applicable system. Consistent with RG 1.206, DAC/ITAAC are not provided for IEEE Std 603, Section 4.2, because the information is provided in the DCD and the IEEE Std 603, Section 4 criteria with DAC/ITAAC adequately verify the design basis information. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, and the staff's safety evaluation for each applicable system provided in this report as part of conformance to 10 CFR 50.55a(h), the staff finds that Section 4.2 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.3, is adequately addressed for the safety systems. This criterion requires the identification of the permissive conditions for each operating bypass capability that is to be provided. The permissive conditions for each operating bypass for each system are discussed in DCD Tier 2, Revision 9, Chapter 7, within the safety evaluation section for each applicable system. Consistent with RG 1.206, DAC/ITAAC are not provided for IEEE Std 603, Section 4.3, because the information is provided in the DCD and the IEEE Std 603, Section 4, criteria with DAC/ITAAC adequately verify the design basis information. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, and the staff's safety evaluation for each applicable system provided in this report as part of conformance to 10 CFR 50.55a(h), the staff finds that Section 4.3 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.4, is adequately addressed for the safety systems. This criterion requires the identification of variables or combinations of variables, or both, that are to be monitored to manually or automatically, or both, control each protective action; the analytical limit associated with each variable, the ranges (normal, abnormal, and accident conditions), and the rates of change of these variables to be accommodated until proper completion of the protective action is ensured. The list of such variables to be monitored is determined as part of the HFE design process described in DCD Chapter 18. The variables that are associated with each event are discussed in the relevant subsection describing the event, as defined in DCD Tier 2, Revision 9, Table 15.1-7. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 4.4 to verify that monitored variables are incorporated into software projects during the software life cycle process. Based on the review of the DCD Tier 2, Revision 9, Chapters 7 and 15 documentation and the review of DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.4 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.5, is adequately addressed for the safety systems. This criterion describes the minimum criteria under which manual initiation and control of protective actions may be allowed. Manual actuation relies on minimum equipment and, once initiated, proceeds to completion unless the operator deliberately intervenes. Failure in the automatic initiation portion of a system-level function does not prevent the manual initiation of the function. In DCD Tier 2, Revision 9, Section 7.1.6.6.1.1, the applicant committed that the software projects' design bases includes (1) the points in time and the plant conditions during which manual control is allowed, (2) the justification for permitting initiation or control subsequent to initiation solely by manual means, (3) the range of environmental conditions imposed upon the operator during normal, abnormal, and accident conditions throughout which the manual operations will be performed, and (4) the variables that will be displayed for the operator to use in taking manual action. DCD Tier 1, Revision 9, Table 2.2.15-2, provides

DAC/ITAAC for IEEE Std 603, Section 4.5, to verify that manual initiation and control information is incorporated into software projects during the software life cycle process. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.5 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.6, is adequately addressed for the safety systems. This criterion requires the identification of the minimum number and location of spatial dependence sensors. These sensors are for those variables in Section 4.4 of IEEE Std 603 that have a spatial dependence. The applicant/licensee's analysis should demonstrate that the number and location of sensors are adequate. The applicant committed in DCD Tier 1, Revision 9, Section 2.2.15, that the software projects' design bases list the minimum number and locations of sensors for those variables that are required to perform a safety function and have a spatial dependence (i.e., variables that depend on position in a particular region). DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 4.6, to verify that information related to the minimum number and location of sensors is incorporated into software projects during the software life cycle process. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.6 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.7, is adequately addressed for the safety systems. This criterion requires the identification of the range of transient and steady-state conditions of both motive and control power and the environment (for example, voltage, frequency, radiation, temperature, humidity, pressure, and vibration) during normal, abnormal, and accident circumstances throughout which the safety system shall perform. DCD Tier 2, Revision 9, Section 7.1.6.6.1.5, states that all Q-DCIS equipment will be environmentally qualified to meet the accident conditions through which it operates to mitigate the consequences of the accident and will be seismically qualified to meet safe-shutdown earthquake levels. DCD Tier 1, Revision 9, Table 3.8.1, specifies the ITAAC for the EQ process for safety mechanical and electrical equipment. The Q-DCIS is powered by four pairs of separate Class 1E ac power supplies. Each Q-DCIS division uses the two independent UPSs from the same division. The applicant committed in DCD Tier 1, Revision 9, Section 2.2.15, that the software projects' design bases list the range of transient and steady state conditions of motive and control power and environment (e.g., voltage, frequency, radiation, temperature, humidity, pressure, and vibration) during normal, abnormal, and accident circumstances throughout which the safety system is to perform. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 4.7, to verify that information related to the range of transient and steady-state conditions is incorporated into software projects during the software life cycle process. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Sections 2.2.15 and 3.2, DAC/ITAAC verification, the staff finds that Section 4.7 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.8, is adequately addressed for the safety systems. This criterion requires the identification of the conditions having the potential for functional degradation of safety system performance and for which provisions shall be incorporated to retain the capability for performing the safety function. Safety mechanical equipment and electrical equipment (which comprises electrical power and I&C equipment) is qualified in accordance with the EQ program described in DCD Tier 2, Revision 9, Sections 3.9 through 3.11. Environmental conditions for the zones where qualified equipment is located are calculated for normal, AOO, test, accident and post accident conditions and are documented in DCD Tier 2, Revision 9, Appendix 3H. DCD Tier 1, Revision 9, Table 2.2.15-2, provides

DAC/ITAAC for IEEE Std 603, Section 4.8, to verify that information related to conditions having the potential for causing functional degradation of safety system performance is incorporated into software projects during the software life cycle process. Based on the review of the DCD Tier 2, Revision 9, Chapter 3, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.8 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.9, is adequately addressed for the safety systems. This criterion requires the identification of the methods used to determine that the reliability of the safety system design is appropriate for each such design. In addition, it requires the identification of the methods used to verify that any qualitative or quantitative reliability goals imposed on the system design are met. DCD Tier 2, Revision 9, Section 7.1.6.6.1.1, states that the Design Reliability Assurance Program (D-RAP) is a program utilized during detailed design and specific equipment selection phases to assure that important ESBWR reliability assumptions of the probabilistic Risk Assessment (PRA) are addressed throughout the plant life. DCD Tier 2, Revision 9, Section 17.4, describes the D-RAP. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 4.9, to verify that information related to the methods used to determine that the reliability of the safety system design is incorporated into software projects during the software life cycle process. Based on the review of DCD Tier 2, Revision 9, Chapters 7 and 17 and DCD Tier 1, Revision 9, Section 2.2.15 regarding DAC/ITAAC verification, the staff finds that the requirements of Section 4.9 of IEEE Std 603 are adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.10, is adequately addressed for the safety systems. This criterion requires the documentation of the critical points in time or the plant conditions, after the onset of a DBE, including (1) the point in time or plant conditions, after the protective actions of the safety system are initiated, (2) the point in time or plant conditions that define the proper completion of the safety function, (3) the point in time or the plant conditions that require automatic control of protective actions, and (4) the point in time or the plant conditions that allow returning a safety system to normal. DCD Tier 2, Revision 9, Section 7.1.2.1.2 and Table 15.1-7 discuss the relevant point in time or plant conditions. DCD Tier 2, Revision 9, Chapter 16, describes the allowable conditions for returning a plant to normal. DCD Tier 1, Revision 9, Table 2.2.15-2 provides DAC/ITAAC for IEEE Std 603, Section 4.10, to verify that information related to the methods used to determine that the reliability of the safety system design is incorporated into software projects during the software life cycle process. Based on the review of the DCD Tier 2, Revision 9, Chapters 7, 15 and 16 documentation, and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.10 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.11, is adequately addressed for the safety systems. This criterion requires the analysis and documentation of any equipment protective provisions that may prevent the safety systems from accomplishing their safety function. The safety systems are designed to accomplish their safety function in accordance with the single failure criteria, IEEE Std 603, Section 5.1. FMEA are performed on the safety system final design. Consistent with RG 1.206, DAC/ITAAC are not provided for IEEE Std 603, Section 4.11 because the DCD provides this information and the IEEE Std 603, Section 4 criteria with DAC/ITAAC adequately verify the design basis information. In addition, DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.1, to verify that the software projects design bases comply with the single failure criterion and that the as-built software projects' test results confirm the results of the FMEA. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 4.11 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 4.12, is adequately addressed for the safety systems. This criterion requires identification of any other special design basis that may be imposed on the system design. DCD Tier 2, Revision 9, Chapter 7, has documented the design bases for each subsystem, including the bases for diversity, interlocks, and regulatory agency criteria. Consistent with RG 1.206, DAC/ITAAC are not provided for IEEE Std 603, Section 4.12 because the DCD provides this information and the IEEE Std 603, Section 4 criteria with DAC/ITAAC adequately verify the design basis information. The staff finds that Section 4.12 of IEEE Std 603 is adequately addressed.

7.1.1.3.10.2 *IEEE Standard 603, Section 5, "Safety - System Criteria"*

The staff evaluated whether IEEE Std 603, Section 5, is adequately addressed using SRP Appendix 7.1-C, Section 5. Section 5 requires that the safety systems, with precision and reliability, maintain plant parameters within acceptable limits established by DBEs.

The staff evaluated whether IEEE Std 603, Section 5.1, is adequately addressed for the safety systems. According to this criterion, no single failure within the safety system shall prevent proper protective action at the system level when required. DCD Tier 2, Revision 9, Sections 7.1.2.3 and 7.1.6.6.1.2, provide a general description of design features that contribute to meeting IEEE Std 603, Section 5.1. These features include (1) the arrangement of the Q-DCIS into four divisions, (2) the redundancy of the intra-divisional and safety to nonsafety fiber optic cable communication paths, (3) the powering of safety cabinets and chassis by redundant safety UPS, and (4) an N-2 design basis. DCD Tier 1, Revision 9, Table 2.2.15-1, documented that in the design each safety platform complies with the requirements of IEEE Std 603, Section 5.1. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.1, for the applicant to perform an analysis, or FMEA, that confirms that the requirements of the single failure criterion are satisfied for the safety systems. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 5.1 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.2, is adequately addressed for the safety systems. This criterion requires the safety system design to provide features to ensure that system-level actions go to completion. DCD Tier 2, Revision 9, Section 7.1.6.6.1.3, provides a general description of design features that contribute to meeting IEEE Std 603, Section 5.2. In accordance with SRP Appendix 7.1-C, the staff's review of this item should include a review of functional and logic diagrams to ensure that seal-in features are provided to enable system-level protective actions to go to completion. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.2, for the applicant to perform an inspection of the design phase summary Baseline Review Report (BRR) of the software project to verify that the seal-in features are provided (DAC requirement). DAC/ITAAC are also provided for the applicant to perform an inspection of the as-built software project installation phase summary BRR to verify that the safety functions of the "execute features" continue until completion. As documented in DCD Tier 2, Revision 9, Section 7.1.6.6.1.3 and Tier 1, Revision 9, Table 2.2.15-2, item (9), the applicant committed in DCD Tier 1, Revision 9, Section 2.2.15, that the I&C platform software projects are designed to include these seal-in features for all safety systems. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 5.2 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.3, is adequately addressed for the safety systems. SRP Appendix 7.1-C states that the applicant should confirm that the quality assurance provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, apply to the safety system. DCD Tier 2, Revision 9, Section 7.1.6.6.1.4, states that the NRC-accepted applicant quality assurance program, with its implementing procedures, constitutes the applicant's quality assurance system that is applied to the Q-DCIS design. Chapter 17 of this report addresses the staff's evaluation of the adequacy of the quality assurance program. In addition, IEEE Std 7-4.3.2, Section 5.3, provides quality requirements for digital computer systems split into six criteria: (1) software development, (2) software tools, (3) verification and validation (V&V), (4) independent V&V requirements, (5) software configuration management, and (6) software risk management. NEDE-33226P and NEDE-33245P describe implementation of these requirements, including the software life cycle process for the safety systems hardware and software. Section 7.1.2.3 of this report provides the staff's evaluation of the applicant's software development activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for the applicant to perform results analyses to confirm that the software development activities for the safety systems are conducted in a manner consistent with the DCD and produce results that satisfy the acceptance criteria in BTP HICB-14. Accordingly, based on the applicant's use of an acceptable software development process, as evaluated in Section 7.1.2.3 of this report, and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.3 of IEEE Std 7-4.3.2 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.4, is adequately addressed for the safety systems. This criterion requires that safety systems be qualified to meet performance requirements identified in the design basis. SRP Appendix 7.1-C provides acceptance criteria for EQ. DCD Tier 2, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Chapter 3 of this report. DCD Tier 2, Revision 9, Sections 7.1.6.4 and 7.1.6.6.1.5, describe how the Q-DCIS components are designed to be qualified by type testing and analysis to perform all safety functions when operated within the specified EMI limits. The Q-DCIS components are qualified in conformance with RG 1.180, when mounted in accordance with the specified methods. The Q-DCIS equipment is designed so that it is not susceptible to electromagnetic disturbances from neighboring modules and will not cause electromagnetic disturbances to neighboring modules. As endorsed by RG 1.180, the EMI qualification design follows the requirements specified in Military (Mil) Std 461E, "Department of Defense Interface Standard - Requirement for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment," and International Electrotechnical Commission (IEC) 61000-4, "International Standard - Electromagnetic Compatibility Testing and Measurement Techniques," depending on the specific requirement conditions. DCD Tier 1, Revision 9, Table 3.8-1, provides ITAAC for the applicant to confirm the EQ of safety electrical and digital I&C equipment, which address the EQ aspects of IEEE Std 603, Section 5.4. Accordingly, based on the applicant's appropriate identification of EQ programs and their confirmation of the ITAAC, the staff finds that the EQ aspects of IEEE Std 603, Section 5.4, is adequately addressed.

In addition, IEEE Std 7-4.3.2, Section 5.4, provides criteria for computer system testing and qualification of existing commercial computers. With regard to computer system testing, IEEE Std 7-4.3.2, Section 5.4.1, states, "Equipment qualification testing shall be performed with the computer functioning with software and diagnostics that are representative of those used in actual operation." In Regulatory Position C(2) of RG 1.209, the staff directly enhanced this statement by adding the following: "The qualification testing should be performed with the I&C system functioning, with software and diagnostics that are representative of those used in actual operation, while the system is subjected to the specified environmental service conditions, including abnormal operational occurrences." In RAI 7.1-47, the staff requested the applicant to demonstrate how this standard is met. In response, the applicant added RG 1.209 to DCD Tier 2, Table 7.1-1, and added discussion of conformance with RG 1.209 in the applicable sections of DCD Tier 2, Chapter 7. The staff finds that the response is acceptable since the applicant revised the DCD to address conformance to RG 1.209. Based on the applicant's response, RAI 7.1-47 is resolved. DCD Tier 1, Revision 9, Table 3.8-1, provides ITAAC for the applicant to confirm the EQ of safety electrical and digital I&C equipment. DCD Tier 1, Revision 9, Table 3.2-1, provides DAC/ITAAC for the applicant to develop and implement software test plans for safety systems. These two sets of DAC/ITAAC together address the computer system testing aspects of IEEE Std 603, Section 5.4. Accordingly, based on the inclusion of IEEE Std 603, Section 5.4, and RG 1.209 commitments in the safety systems design basis and verification of the EQ and the inclusion of computer system testing in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the computer system testing aspects of IEEE Std 603, Section 5.4, are adequately addressed.

With regard to the qualification of existing commercial computers, IEEE Std 7-4.3.2, Section 5.4.2, provides criteria for the qualification of existing commercial computers. The NRC has approved the use of two EPRI reports for conforming to this criterion; EPRI TR-106439, as approved by the NRC in its safety evaluation dated July 17, 1997, and EPRI TR-107330, as approved by the NRC on July 30, 1998. In DCD Tier 2, Revision 9, Section 7.2.1.3.5, the applicant states that the Q-DCIS hardware, embedded and operating system software, and peripheral components conform to the guidance in BTP HICB-18, "Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems." The Q-DCIS is built and qualified specifically for ESBWR applications as safety, rather than commercial-grade, PLCs. The Q-DCIS meets the acceptance criteria contained in BTP HICB-14, for safety applications. NEDE-33226P, Section 5.8.3.6, states that commercial off the shelf (COTS) software to be used in a safety application shall be dedicated in accordance with an NRC-acceptable method of commercial-grade dedication for software and digital components (e.g., EPRI TR-106439). NEDE-33226P, Section 5.8.3.6, provides an additional description of the qualification of existing commercial computers for software. In its description of the software development plan (SDP), NEDE-33226P states that the COTS Evaluation Report and Documentation Package will be a design-phase output document. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm the implementation of the SDP. Accordingly, the staff finds that the criterion for the qualification of existing commercial computers is adequately addressed for the DCIS. Based on the applicant's commitment to use NRC-acceptable qualification methods and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the qualification of existing commercial computers in IEEE Std 603, Section 5.4, is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.5, is adequately addressed for the safety systems. This criterion requires that the safety system accomplish its safety functions under the full range of applicable conditions enumerated in the design basis. For digital computer-based systems, system real-time performance should be adequate to ensure completion of protective actions within the critical points of time identified, as required by IEEE Std 603, Section 4.10. BTP HICB-21 provides supplemental guidance on evaluating response times for digital computer-based systems and discusses design constraints that allow greater confidence in the results analyses or prototype testing to determine real-time performance. The complete design basis for real-time performance is not available in the DCD. In the response to RAI 7.9-10, the applicant stated, “NEDE-33226P and NEDE-33245P define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested.” DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to verify activities associated with NEDE-33226P and NEDE-33245P. DCD Tier 1, Revision 9, Table 3.8-1, provides ITAAC for the applicant to confirm the EQ of safety electrical and digital I&C equipment, which confirm that the safety systems function in the full range of applicable conditions enumerated in the design basis consistent with IEEE Std 603, Section 5.5.

IEEE Std 7-4.3.2 indicates that designs for computer system integrity and for test and calibration should be addressed as part of safety system integrity. SRP Appendix 7.1-D states that computer system software integrity (including the effects of hardware-software interaction) should be demonstrated by the applicant’s software safety analysis activities. BTP HICB-14, Section B.3.1.9, describes the acceptable characteristics of software safety plans. BTP HICB-14, Section B.3.2.1, describes the characteristics of acceptable software safety analyses. As mentioned above for the plant performance requirements, DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to verify the software safety plans and analyses for the RPS. The staff finds that IEEE Std 7-4.3.2 is adequately addressed.

SRP Appendix 7.1-D states that the design should provide for safety systems to fail in a safe state, or into a state that is demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy, or adverse environments are experienced. This aspect is typically addressed by the applicant’s FMEA. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.1, for the applicant to perform an analysis, or FMEA, that confirms that the requirements of the single failure criterion are satisfied for the safety systems. The staff finds that these FMEAs are an acceptable method of addressing this aspect of IEEE Std 603, Section 5.5.

Accordingly, based on the inclusion of IEEE Std 603, Section 5.5, in the safety systems design basis and its confirmation in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.5 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.6, is adequately addressed for the safety systems. This criterion requires, in part, independence among (1) redundant portions of a safety system, (2) safety systems and the effects of DBEs, and (3) safety systems and other systems. The following three aspects of independence should be addressed in each case:

(1) Physical independence
(2) Electrical independence
(3) Communications independence

DCD Tier 2, Section 7.1.2.4, specifies conformance to RG 1.75 and IEEE Std 384. RG 1.75 and IEEE Std 384 provide criteria for the independence of electrical safety systems, including physical separation and electrical isolation. Section 7.1.5 of this report evaluates communications independence. DCD Tier 2, Revision 9, Sections 7.1.2.1.1 and 7.1.6.6.1.7, provides a general description of the design features that contribute to meeting IEEE Std 603, Section 5.6. These features include: (1) arrangement of the Q-DCIS into four redundant and independent divisions, (2) provision of an independent electrical power source for each division, and (3) design of the sensors for each division to be independent and physically separated. DCD Tier 2, Revision 9, Table 7.1-2, identifies DCD sections in which IEEE Std 603, Section 5.6, is addressed for specific systems, which are evaluated throughout this report.

DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.6, to verify that design features that provide physical, electrical, and communications independence are incorporated into software projects during the software life cycle process. DCD Tier 1, Table 2.2.15-2, also provides DAC/ITAAC for IEEE Std 603, Section 5.6, to verify that as-built software projects have (1) four independent redundant divisions, (2) communications independence, and (3) independence between safety systems and nonsafety equipment. Accordingly, based on the inclusion of IEEE Std 603, Section 5.6, in the safety systems design basis and the review of DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 5.6 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.7, is adequately addressed for safety systems. This criterion requires that the capability for testing and calibration of safety system equipment be provided while retaining the capability of the safety systems to accomplish their safety functions. DCD Tier 2, Section 7.1.6.6.1.8, provides a general description of design features that contribute to meeting IEEE Std 603, Section 5.7. DCD Tier 2, Table 7.1-2, identifies DCD sections in which IEEE Std 603, Section 5.7, is addressed for specific systems. The corresponding sections of this report evaluate these discussions. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.7, to verify that (1) maintenance bypasses allow test and calibration of one out of four divisions, (2) the divisions not in bypass status will accomplish their safety functions, (3) bypassed divisions alarm in the MCR, and (4) the division logic automatically becomes a two-out-of-three voting scheme. SRP Appendix 7.1-C also states that any failure that is not detectable must be considered concurrently with any random postulated, detectable, single failure. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.1, for the applicant to perform an analysis, or FMEA, which confirms that the requirements of the single failure criterion are satisfied for the safety systems. Accordingly, based on the inclusion of IEEE Std 603, Section 5.7, in the safety systems' design basis and its verification in the DCD Tier 1 Revision 9, DAC/ITAAC, the staff finds that Section 5.7 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.8, is adequately addressed for the safety systems. This criterion is associated with information displays and inoperable surveillance. DCD Tier 2, Section 7.1.6.6.1.9, provides a general description of design features that contribute to meeting IEEE Std 603, Section 5.8. DCD Tier 2, Table 7.1-2, identifies DCD sections in which IEEE Std 603, Section 5.8, is addressed for specific systems, which are evaluated throughout this report. SRP Appendix 7.1-C states that safety system bypass and inoperable status indications should conform to RG 1.47. DCD Tier 2, Revision 9, Section 7.1.6.4, specifies that bypass indications are designed to satisfy RG 1.47. DCD Tier 2, Revision 9, Chapter 18, which is evaluated in Chapter 18 of this report, describes the HFE design process to design information displays. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification applies to all safety systems and includes verifying the inventory of displays for manually controlled actions, system status indications, and indications of bypasses. Accordingly, based on the inclusion of IEEE Std 603, Section 5.8, in the safety systems' design basis and its confirmation in the DCD Tier 1 Revision 9, DAC/ITAAC, the staff finds that Section 5.8 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.9, is adequately addressed for the safety systems. This criterion requires that the design permit the administrative control of access to safety system equipment. This criterion also requires that the administrative controls be supported by provisions within the safety systems, by provisions in the generating station design, or by a combination thereof. DCD Tier 2, Section 7.1.6.6.1.11, generally describes access controls to safety I&C systems. Keys, passwords, and other security devices are used to control access to specific rooms; open specific equipment cabinets; obtain permission to access specific electronic instruments for calibration, testing, and setpoint changes; and gain access to safety system software and data. DCD Tier 2, Revision 9, Section 13.6.1.1.5, also describes access controls for certain I&C cabinets. DCD Tier 1, Revision 9, Section 2.2.15, provides DAC/ITAAC for IEEE Std 603. DCD Tier 2, Revision 9, Section 7.1.6.6.1.10, states that computer-related access controls and authorization are part of the cyber-security program plan, which is described in NEDO-33295 and NEDE-33295P. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm the implementation of the cyber security program. Based on the review of the DCD Tier 2, Revision 9, Chapter 7, documentation and DCD Tier 1, Revision 9, DAC/ITAAC verification, the staff finds that Section 5.9 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.10, is adequately addressed for the safety systems. This criterion requires that the safety systems be designed to facilitate timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment. SRP Appendix 7.1-C states that digital safety systems may include self-diagnostic capabilities to aid in troubleshooting, but the use of self-diagnostics does not replace the need for the capability for test and calibration systems, as required by IEEE Std 603, Sections 5.7 and 6.5. DCD Tier 2, Revision 9, Section 7.1.6.6.1.11, specifies that the Q-DCIS provide periodic self-diagnostic functions to locate failures to the component level. DCD Tier 2, Section 7.1.6.6.1.11, also specifies that the Q-DCIS provide, through the ability to bypass individual divisions, the capability to repair or replace a failed component online without affecting the safety system protection function. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.10, to verify that the software projects have self-diagnostic features that facilitate the timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment. Accordingly, based on the inclusion of IEEE Std 603, Section 5.10, in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.10 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.11, is adequately addressed for the safety systems. This criterion requires the following:

- Safety system equipment should be distinctly identified for each redundant portion of a safety system in accordance with the requirements of IEEE Std 384.

- Components or modules mounted in equipment or assemblies that are clearly identified as being in a single redundant portion of a safety system should not themselves require identification.

- The identification of safety system equipment should be distinguishable from other purposes.

- The identification of safety system equipment should not require frequent use of reference material.

- The associated documentation should be distinctly identified.

SRP Appendix 7.1-A states that RG 1.75, which endorses IEEE Std 384, provides guidance on identification. The preferred identification method is color coding of components, cables, and cabinets. DCD Tier 2, Revision 9, Section 7.1.2.4, states that the Q-DCIS conforms to RG 1.75 and IEEE Std 384.

DCD Tier 2, Revision 9, Section 7.1.6.6.1.12, provides a general description of conformance to IEEE Std 603, Section 5.11, including a statement that color coding is used as a method of identification and safety equipment is distinctly marked in each redundant division of safety systems. DCD Tier 2, Revision 9, Section 8.3.1.3 also identifies that the identification method is color coding and specifies additional methods of identification, including:, (1) all markers within a division have the same color, (2) the ESBWR standard plant design eliminates safety associated circuits as defined by IEEE Std 384 and in accordance with RG 1.75, (3) divisional separation requirements of individual pieces of hardware are shown in the system elementary diagrams, and (4) identification of raceways, cables, and the like is compatible with the identification of the safety equipment with which it interfaces. DCD Tier 2, Revision 9, Section 8.3.1.3, details how identification will be implemented for equipment, cables, and raceways. The staff finds that the applicant adequately commits to implementing the requirements of IEEE Std 603, Section 5.11. DCD Tier 1, Revision 9, Section 2.2.15, provides DAC/ITAAC for IEEE Std 603, Section 5.11, to verify the distinct identification of each redundant portion of safety systems.

In addition, IEEE Std 7-4.3.2, Section 5.11, provides additional criteria for computer system testing and qualification of existing commercial computers, including (1) firmware and software identification shall be used to assure that the correct software is installed in the correct hardware component and (2) means shall be included in the software such that the identification may be retrieved from the firmware using software maintenance tools. NEDE-33245P, Section 6.4.1, specifies guidelines for configuration identification consistent with IEEE Std 7-4.3.2, Section 5.11, as part of the software configuration management. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to verify activities associated with NEDE-33226P and NEDE-33245P. Accordingly, based on the inclusion of IEEE Std 603, Section 5.11, and IEEE Std 7-4.3.2, Section 5.11, in the safety systems design basis and their confirmation in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.11 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.12, is adequately addressed for the safety systems. This criterion states the following:

- Auxiliary supporting features shall meet all requirements of this standard.

- Other auxiliary features that perform a function that is not required for the safety systems to accomplish their safety functions, or are part of the safety system by association, shall be designed to meet those criteria necessary to ensure that such components, equipment, and systems do not degrade the safety systems below an acceptable level.

DCD Tier 2, Revision 9, Section 7.1.6.6.1.13, identifies two auxiliary systems applicable to the power systems and HVAC. DCD Tier 2, Revision 9, Section 8.3, identifies that separate ac and dc power systems are provided for safety and nonsafety systems. Section 8.3 of this report evaluates these systems and finds that there is acceptable physical separation and independence between the safety and nonsafety systems, including the auxiliary systems.

DCD Tier 2, Revision 9, Section 7.1.6.6.1.13, states that, if the nonsafety redundant HVAC is not available, safety temperature sensors with 2/4 logic trip the control room power that feeds pre-defined components of the nonsafety I&C and other pre-defined nonsafety heat loads. DCD Tier 2, Revision 9, Sections 6.4.4, 9.4.1, 9.4.1.1, and 9.4.1.2, clarify that the purpose of this trip is to remove the heat load caused by the N-DCIS. DCD Tier 2, Section 7.1.6.6.1.13, states that the Q-DCIS and support equipment is qualified for the expected temperature rise; therefore, the Q-DCIS depends on the trip of the N-DCIS to minimize the temperatures in the safety I&C rooms. In RAI 7.1-132, the staff requested that the applicant include a discussion of this trip in the applicable instrumentation sections of the system descriptions in the DCD. In response to RAI 7.1-132, which was incorporated in DCD Revision 6, the applicant updated DCD Tier 2, Sections 6.4.8, 7.3.4.2, and 9.4.1.5, to clarify the function of the safety CRHAVS emergency trip circuit for the N-DCIS equipment installed in the CRHA. The applicant updated DCD Tier 1, Tables 2.2.13-2 and 2.2.13-3 for ITAAC verification. The staff finds the response is acceptable since the applicant clarified the trip of the N-DCIS throughout the DCD. Based on the applicant's response, this aspect of RAI 7.1-132 is resolved. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.12, to verify that auxiliary features do not degrade the performance of software projects. Accordingly, based on the inclusion of IEEE Std 603, Section 5.12, in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.12 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.13, is adequately addressed for the safety systems. This criterion states that the sharing of SSCs between units at multi-unit generating stations is permissible, provided that the ability to simultaneously perform required safety functions in all units is not impaired. In RAI 7.1-134, the staff requested that the applicant clarify whether there are shared components, and if so, clarify why the failure of shared components does not impact the Q-DCIS. DCD Tier 2, Revision 5, Section 7.1.6.6.1.14, states that for multiple unit designs only the N-DCIS would have common network components necessary to control and monitor common hardware and systems. DCD Tier 2, Revision 5, Section 7.1.6.6.1.14, also states that the operation or failure of shared N-DCIS components does not affect the performance of the Q-DCIS. RAI 7.1-134 was being tracked as an open item in the SER with open items. In response, the applicant revised DCD Tier 2, Section 7.1.6.6.1.14, to state that the multi-unit station criteria do not apply to the standard single unit plant design submitted for NRC certification and to remove statements concerning multiple units in DCD Tier 1 and Tier 2. The staff finds the response is acceptable, since the applicant removed references to a multi-unit station from the DCD. Based on the applicant's response, RAI 7.1-134 is resolved. The staff finds that IEEE Std 603, Section 5.13, does not apply to design certification.

The staff evaluated whether IEEE Std 603, Section 5.14, is adequately addressed for the safety systems. This criterion states that human factors shall be considered at the initial stages, and throughout the design process, to ensure that the functions allocated in whole or in part to the human operator or operators and maintenance personnel can be successfully accomplished to meet the safety system's design goals. The I&C design is integrated with the HFE design process, as described in DCD Tier 2, Chapter 18, and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. Accordingly, based on the inclusion of IEEE Std 603, Section 5.14, in the safety systems' design basis and its verification in the HFE DAC/ITAAC, the staff finds that Section 5.14 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 5.15, is adequately addressed. This criterion requires performing an analysis of the design to confirm that established reliability goals are achieved. SRP Appendix 7.1-C states that the applicant should justify that the degree of redundancy, diversity, testability, and quality provided in the safety system design is adequate to achieve functional reliability commensurate with the safety functions to be performed. SRP Appendix 7.1-C further states that the staff considers software that complies with the quality criteria of IEEE Std 603, Section 5.3, and is used in safety systems that provide measures for defense against CCFs, as described in the SRP Appendix 7.1-C discussion of IEEE Std 603, Section 5.1, complies with the fundamental reliability requirements of IEEE Std 603. DCD Tier 1, Revision 9, Section 3.2, provides DAC/ITAAC for the applicant to perform results analyses to confirm that the software development activities for the safety systems are conducted consistently with the DCD and produce results that satisfy the acceptance criteria in BTP HICB-14, consistent with IEEE Std 603, Section 5.3. DCD Tier 1, Revision 9, Section 2.2.15, includes the DAC/ITAAC for IEEE Std 603, Sections 5.1 to verify the implementation of the single failure criteria. DCD Tier 2, Revision 9, Section 7.1.6.6.1.16, identifies that the D-RAP, evaluated in Section 17.4 of this report, confirms that any quantitative or qualitative reliability goals established for the protection systems are met. DCD Tier 1, Revision 9, Section 3.6, provides ITAAC to confirm the reliability of the SSCs in the D-RAP, including the software projects. Accordingly, based on the inclusion of IEEE Std 603, Section 5.15, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.15 is adequately addressed.

7.1.1.3.10.3 *IEEE Standard 603, Section 6, "Sense and Command Features - Functional and Design Requirements"*

The staff evaluated whether IEEE Std 603, Section 6.1, is adequately addressed. This criterion states that means shall be provided to automatically initiate and control all protective actions except as justified in Section 4.5. SRP Appendix 7.1-C states that (1) the applicant's analysis should confirm that the safety system is qualified to demonstrate that the performance requirements are met, (2) the evaluation of the precision of the safety system should be addressed to the extent that setpoints, margins, errors, and response times are factored into the analysis, (3) for digital computer-based systems, the evaluation should confirm that the functional requirements are appropriately allocated into hardware and software requirements, and (4) the evaluation should also confirm that the system's real-time performance is deterministic and known. DCD Tier 2, Revision 9, Sections 7.1.2.1.2 and 7.1.6.6.1.17, provide a broad description of the design basis applicable to this criterion. NEDE-33226P and NEDE-33245P address the first three SRP topics by defining a software development process by which plant performance requirements for I&C systems under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to verify activities associated with NEDE-33226P and NEDE-33245P. DCD Tier 2, Revision 9, Section 7.1.3.2.7, addresses the fourth topic by specifying that the Q-DCIS internal and external communication protocols are deterministic. DCD Tier 1, Revision 9, Table 2.2.15-2, provides the DAC/ITAAC for IEEE Std 603, Section 6.1, to verify that information related to the methods used to determine that the reliability of the safety system design is incorporated into software projects during the software life cycle process. Accordingly, based on the inclusion of IEEE Std 603, Section 6.1, in the safety systems design basis and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 6.1 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 6.2, is adequately addressed. This criterion requires, in part, that (1) means shall be provided in the control room to implement manual initiation at the division level of the automatically initiated protective actions, (2) means shall be provided in the control room to implement manual initiation and control of the protective actions identified in IEEE Std 603, Section 4.5, that have not been selected for automatic control under IEEE Std 603, Section 6.1, and (3) means shall be provided to implement the manual actions necessary to maintain safe conditions after the protective actions are completed as specified in IEEE Std 603, Section 4.10. SRP Appendix 7.1-C states that features for manual initiation of protective action should conform to RG 1.62. DCD Tier 2, Revision 9, Section 7.1.6.6.1.18, states, “Each protective action can be manually initiated at the system level, in conformance to RG 1.62, and at the division level in conformance to IEEE Std 603, Sections 6.2 and 7.2.” DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with IEEE Std 603, Section 6.2. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.2, to confirm that applicable systems have MCR features that are capable of manually initiating and controlling automatically initiated safety functions at the division level. DCD Tier 2, Section 7.1.6.6.1.18, also identifies that the design of manual control is integrated into the overall HFE design as described in DCD Tier 2, Revision 9, Chapter 18, and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for verifying the implementation of the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. Accordingly, based on the inclusion of IEEE Std 603, Section 6.2, in the safety systems design basis and its verification in the DCD Tier 1 Revision 9, DAC/ITAAC, the staff finds that Section 6.2 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 6.3, is adequately addressed. This criterion requires that any failure of nonsafety systems should not affect safety protection systems or prevent them from performing their safety functions. For example, DI&C-ISG-04, Revision 1, “Interim Staff Guidance on Highly-Integrated Control Rooms—Communications Issues (HICRc),” states that communication faults should not adversely affect the performance of required safety functions in any way. Faults, including communication faults, originating in nonsafety equipment, do not constitute “single failures,” as described in the single failure criterion of Appendix A to 10 CFR Part 50 (see GDC 24). DCD Tier 2, Appendix 7.1-C states that, for those cases in which the event of concern is the single failure of a sensing channel shared between control and protection functions, previously accepted approaches have included the following:

- Isolating the safety system from channel failure by providing additional redundancy

- Isolating the control system from channel failure by using data validation techniques to select a valid control input

DCD Tier 2, Section 7.1.6.6.1.19, provides a general description of conformance by stating, “The Q-DCIS protection systems are separate and independent from the nonsafety control systems.” DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.3, to verify that software projects have (1) four independent redundant division, (2) communications independence, (3) independence between safety systems and nonsafety equipment. Accordingly, based on the inclusion of IEEE Std 603, Section 6.3, in the safety systems design basis and its verification in the DCD Tier 1 Revision 9, DAC/ITAAC, the staff finds that Section 6.3 is adequately addressed.

The staff evaluated whether Section 6.4 of IEEE Std 603 is adequately addressed. This criterion states that, to the extent feasible and practical, sense and command feature inputs shall be derived from signals that are direct measures of the desired variables, as specified in the design basis. SRP Appendix 7.1-C states that the applicant should verify that any indirect parameter is a valid representation of the desired direct parameter for all events. DCD Tier 2, Section 7.1.6.6.1.20, states, “To the extent feasible, the protection system inputs are derived from signals that directly measure the designated process variables,” which is consistent with the high level conceptual nature of the DCIS design. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.4, to confirm that sense and command feature inputs for software projects are derived from signals that are direct measures of the desired variables specified in the design bases.

SRP Appendix 7.1-C states that, for both direct and indirect parameters, the applicant should verify that the characteristics (e.g., range, accuracy, resolution, response time, sample rate) of the instruments that produce the safety system inputs are consistent with the analysis provided in DCD Tier 2, Chapter 15. NEDE-33226P defines the software development process by which plant performance requirements for I&C systems under various operational conditions will be specified, implemented, and undergo software functional testing. NEDE-33245 defines the process for conducting software safety analysis, verification and validation activities, and testing (i.e., validation and acceptance testing: software validation test [SVT], system factory acceptance test [SFAT], multi-system factory acceptance test [MFAT], and site acceptance test [SAT]) during the software development process to ensure that plant performance requirements for I&C systems under various operational conditions are satisfied. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to verify activities associated with NEDE-33226P and NEDE-33245P. Accordingly, based on the inclusion of IEEE Std 603, Section 6.4, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 6.4 is adequately addressed.

The staff evaluated whether Section 6.5 of IEEE Std 603 is adequately addressed. This criterion states, in part, that means shall be provided for checking the operational availability of each sensor required for a safety function. SRP Appendix 7.1-C provides guidance on the checking of sensors. DCD Tier 2, Revision 9, Section 7.1.6.6.1.21, states that protection system sensors have the capability to be checked by perturbing the monitored variable, by varying the input to the sensor within the constraints, or by cross-checking between redundant channels. This capability is consistent with SRP Appendix 7.1-C. DCD Tier 2, Section 7.1.6.6.1.21, also describes features to provide at least two valid divisions for cross-checking of monitored variables. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.5, to verify that (1) maintenance bypasses allow test and calibration of one out of four divisions, (2) the divisions not in bypass status will accomplish their safety functions, (3) bypassed divisions alarm in the MCR, and (4) the division logic automatically becomes a two-out-of-three voting scheme. Accordingly, based on the inclusion of IEEE Std 603, Section 6.5, in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 6.5 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 6.6, is adequately addressed. This criterion states, in part, that whenever the applicable permissive conditions are not met, a safety system shall automatically prevent the activation of an operating bypass or initiate the appropriate safety function or functions. SRP Appendix 7.1-C states that the operator may take action to prevent the unnecessary initiation of a protective action. DCD Tier 2, Revision 9, Section 7.1.6.6.1.22, describes the applicability of IEEE Std 603, Section 6.6, to the protection systems and states that the design provides for the automatic removal of operational bypasses. DCD Tier 2, Revision 9, Sections 7.2.1.5 and 7.3.5.2, describe the Q-DCIS operating bypasses and provisions for their automatic removal. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with IEEE Std 603, Section 6.6 criterion. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.6, to verify that software projects are capable of automatically (1) preventing the activation of an operating bypass whenever the applicable permissive conditions for an operating bypass are not met and (2) removing activated operating bypasses if the plant conditions change so that an activated operating bypass is no longer permissible. Accordingly, based on the inclusion of IEEE Std 603, Section 6.6, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 6.6 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 6.7, is adequately addressed. This criterion states, in part, that the capability of a safety system to accomplish its safety function shall be retained while sense and command features equipment is in maintenance bypass. The criterion further states that, during such operation, the sense and command features shall continue to meet the requirements of Sections 5.1 and 6.3. DCD Tier 2, Section 7.1.6.6.1.23, describes the general capability of safety systems to accomplish their safety functions while a safety division is in maintenance bypass. DCD Tier 2, Section 7.1.6.6.1.23, states that this capability is provided since only one safety division, out of four, may be bypassed at any given time. DCD Tier 2, Revision 9, Sections 7.2.1.5.2.2 and 7.3.5.2.4, describe the ability of the Q-DCIS systems to accomplish their safety functions while a safety division is in maintenance bypass. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with IEEE Std 603, Section 6.7, criterion. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.7, to verify that software projects are capable of performing their safety functions when one division is in maintenance bypass. Accordingly, based on the inclusion of IEEE Std 603, Section 6.7, in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 6.7 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 6.8, is adequately addressed. This criterion states that the allowance for uncertainties between the process analytical limit documented in IEEE Std 603, Section 4.4, and the device setpoint shall be determined using a documented methodology. The criterion also states that, where it is necessary to provide multiple setpoints for adequate protection for a particular mode of operation or set of operating conditions, the design shall provide a positive means of ensuring that the more restrictive setpoint is used when required. DCD Tier 2, Revision 9, Section 7.1.6.6.1.24, states that instrument setpoints are determined by the methodology described in NEDO-33304, which is evaluated in Section 7.1.4 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 6.8, to confirm that the safety systems' setpoints for safety functions are defined, determined, and implemented based on a defined setpoint methodology. Accordingly, based on the inclusion of IEEE Std 603, Section 6.8, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 6.8 is adequately addressed.

7.1.1.3.10.4 *IEEE Standard 603, Section 7, "Execute Features - Functional and Design Requirements"*

The staff evaluated whether IEEE Std 603, Section 7.1, is adequately addressed. Section 7.1 states, in part, that the safety system should, with precision and reliability, automatically initiate and execute protective action for the range of conditions and performance requirements. SRP Appendix 7.1-C states that (1) the applicant's analysis should confirm that the safety system is qualified to demonstrate that the performance requirements are met, (2) the evaluation of the precision of the safety system should be addressed to the extent that setpoints, margins, errors, and response times are factored into the analysis, (3) for digital computer-based systems, the evaluation should confirm that the functional requirements are appropriately allocated into hardware and software requirements, and (4) the evaluation should also confirm that the system's real-time performance is deterministic and known. DCD Tier 2, Section 7.1.6.6.1.17, provides a broad description of the design basis applicable to this criterion. NEDE-33226P and

NEDE-33245P address the first three SRP topics by defining a software development process by which plant performance requirements for I&C systems under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to verify activities associated with NEDE-33226P and NEDE-33245P. DCD Tier 2, Revision 9, Section 7.1.3.2.7, addresses the fourth topic by specifying that the Q-DCIS internal and external communication protocols are deterministic. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with the IEEE Std 603, Section 7.1, criterion. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 7.1, to verify that design features that support deterministic performance are incorporated into software projects during the software life cycle process. DCD Tier 1, Table 2.2.15-2, also provides DAC/ITAAC for IEEE Std 603, Section 7.1, to confirm the automatic initiation and control of the required safety functions. Accordingly, based on the inclusion of IEEE Std 603, Section 7.1, in the safety systems' design basis and DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC verification, the staff finds that Section 7.1 of IEEE Std 603 is adequately addressed.

The staff reviewed whether Section 7.2 of IEEE Std 603 is adequately addressed. Section 7.2 states, in part, that the review of manual controls should confirm that the controls will be functional (e.g., power will be available and command equipment is appropriately qualified) and accessible within the time required of the operator during plant conditions under which manual actions may be necessary. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with IEEE Std 603, Section 7.2. DCD Tier 1, Revision 9, Table 2.2.15-2, provides the DAC/ITAAC for IEEE Std 603, Section 7.2, to confirm that applicable systems have MCR features capable of manually initiating and controlling automatically initiated safety functions at the division level. DCD Tier 2, Revision 9, Section 7.1.6.6.1.18, identifies that the design of manual control is integrated into the overall HFE design as described in DCD Tier 2, Revision 9, Chapter 18, which is evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for verifying the implementation of the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. Accordingly, based on the inclusion of IEEE Std 603, Section 7.2, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 7.2 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 7.3, is adequately addressed. This criterion requires that the design of the execute features shall be such that once initiated, the protective actions of the execute features shall go to completion. DCD Tier 2, Revision 9, Section 7.1.6.6.1.3, provides a general description of design features that contribute to meeting IEEE Std 603, Section 7.3. In accordance with SRP Appendix 7.1-C, the staff review of this item should include a review of functional and logic diagrams, which are not available at this time, to ensure that seal-in features are provided to enable system-level protective actions to go to completion. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with IEEE Std 603, Section 7.3. DCD Tier 1, Revision 9, Table 2.2.15-2 provides the DAC/ITAAC for IEEE Std 603, Section 7.3, for the applicant to perform an inspection of the design phase summary BRR of the software project to verify that the seal-in features are provided (DAC requirement). DAC/ITAAC are also provided for the applicant to perform an inspection of the as-built software project installation phase summary BRR to verify that the safety functions of the "execute features" continue until completion. As documented in DCD Tier 1, Revision 9, Table 2.2-15 and DCD Tier 2, Section 7.1.6.6.1.3, the applicant committed that the I&C platform software projects are designed to include these seal-in feature for all safety systems. Accordingly, based on the inclusion of IEEE Std 603, Section 7.3, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 7.3 of IEEE Std 603 is adequately addressed.

The staff reviewed whether Section 7.4 of IEEE Std 603 is adequately addressed. This requirement states, in part, that whenever the applicable permissive conditions are not met, a safety system shall automatically prevent the activation of an operating bypass or initiate the appropriate safety function or functions. SRP Appendix 7.1-C states that the operator may take action to prevent the unnecessary initiation of a protective action. DCD Tier 2, Revision 9, Section 7.1.6.6.1.22, describes the applicability of IEEE Std 603, Section 7.4, to the protection systems and states that the design provides for the automatic removal of operational bypasses. DCD Tier 2, Revision 9, Sections 7.2.1.5 and 7.3.5.2, which are evaluated in Sections 7.2 and 7.3 of this report, respectively, describe the Q-DCIS operating bypasses and provisions for their automatic removal. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with IEEE Std 603, Section 7.4 criterion. DCD Tier 1, Revision 9, Table 2.2.15-2, provides the DAC/ITAAC for IEEE Std 603, Section 7.4, to verify that software projects are capable of automatically (1) preventing the activation of an operating bypass whenever the applicable permissive conditions for an operating bypass are not met and (2) removing activated operating bypasses if the plant conditions change so that an activated operating bypass is no longer permissible. Accordingly, based on the inclusion of IEEE Std 603, Section 7.4, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 7.4 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 7.5, is adequately addressed. This criterion states that the capability of a safety system to accomplish its safety function shall be retained while the execute features equipment is in maintenance bypass. DCD Tier 2, Revision 9, Section 7.1.6.6.1.23, describes the general capability of safety systems to accomplish their safety functions while a safety division is in maintenance bypass. DCD Tier 2, Section 7.1.6.6.1.23, states that this capability is provided since only one safety division, out of four, may be bypassed at any given time. DCD Tier 2, Revision 9, Sections 7.2.1.5 and 7.3.5.2, which are evaluated in Sections 7.2 and 7.3 of this report, respectively, describe the ability of the Q-DCIS systems capability to accomplish their safety functions while a safety division is in maintenance bypass. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with the IEEE Std 603, Section 7.5, criterion. DCD Tier 1, Revision 9, Table 2.2.15-2, provides the DAC/ITAAC for IEEE Std 603, Section 7.5, to verify that software projects are capable of performing their safety functions, when one division is in maintenance bypass. Accordingly, based on the inclusion of IEEE Std 603, Section 7.5, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 7.5 of IEEE Std 603 is adequately addressed.

7.1.1.3.10.5 *IEEE Standard 603, Section 8, "Power Source Requirements"*

The staff evaluated whether IEEE Std 603, Section 8.1, is adequately addressed. This criterion states that those portions of the Class 1E power system that are required to provide power to the many facets of the safety system are governed by the criteria of IEEE Std 603 and are a part of the safety systems. The criterion further states that IEEE Std 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," provides specific criteria unique to the Class 1E power systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.25, states that the Q-DCIS protection system cabinets and components are supported by two independent power sources. Each division of safety I&C is powered by two UPSs that can supply 120 volts ac from offsite power, diesel generator power, or safety batteries (for 72 hours). Either of the two power sources allows the Q-DCIS operation. DCD Tier 1, Revision 9,

Table 2.2.15-2, provides the DAC/ITAAC for IEEE Std 603, Section 8.1, to verify that the software project's electrical components receive power from their respective, divisional, safety-related power supplies. Accordingly, based on the inclusion of IEEE Std 603, Section 8.1, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 8.1 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 8.2, is adequately addressed. This criterion states that power sources, such as control air systems, bottled gas systems, and hydraulic systems, required to provide power to the safety systems are a part of the safety systems and shall provide power consistent with the requirements of IEEE Std 603. DCD Tier 2, Revision 9, Section 7.1.6.6.1.26, states that if a non-electrical power source is required for a safety function, then the source of the power is classified as safety-related and complies with IEEE Std 603. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with the IEEE Std 603, Section 8.2, criterion. DCD Tier 1, Revision 9, Table 2.2.15-2 provides the DAC/ITAAC for IEEE Std 603, Section 8.2, to verify that the software project's safety systems and components that require non-electric power receive it from safety-related sources. Accordingly, based on the inclusion of IEEE Std 603, Section 8.2, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 8.2 of IEEE Std 603 is adequately addressed.

The staff evaluated whether IEEE Std 603, Section 8.3, is adequately addressed. This criterion states that the capability of the safety systems to accomplish their safety functions shall be retained while power sources are in maintenance bypass. The criterion further states that portions of the power sources with a degree of redundancy of one shall be designed such that, when a portion is placed in maintenance bypass (i.e., reducing temporarily its degree of redundancy to zero), the remaining portions provide acceptable reliability. DCD Tier 2, Revision 9, Section 7.1.6.6.1.27, states that the Q-DCIS components are powered by redundant, independent, and separated UPS appropriate to their division with battery backup (per division) for at least 72 hours. Operation of the Q-DCIS when one of its power supplies is in maintenance bypass technically makes one division inoperable since the maintenance bypass reduces a division's ability to operate to approximately 36 hours (from 72 hours) should offsite or diesel power be lost. However, since the Q-DCIS retains full functionality with three out of four divisions operable, the staff finds the reduced operating lifetime of a single division upon loss of power acceptable. DCD Tier 1, Revision 9, Table 2.2.15-1, requires that all safety I&C platforms comply with the IEEE Std 603, Section 8.3, criterion. DCD Tier 1, Revision 9, Table 2.2.15-2 provides the ITAAC for IEEE Std 603, Section 8.3, to verify that software projects are capable of performing their safety functions, when one power supply division is in maintenance bypass. Accordingly, based on the inclusion of IEEE Std 603, Section 8.3, in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 8.3 of IEEE Std 603 is adequately addressed.

7.1.1.4 *Conclusion*

Based on the above, and additional details provided in Sections 7.1.1.3.1 through 7.1.1.3.10 of this report, the staff concludes that the applicant has identified the I&C systems that are important to safety. The applicant has identified the NRC regulations that are applicable to these systems. The applicant has also identified appropriate guidelines consisting of the regulatory guides and the industry codes and standards applicable to the systems. The staff concludes that the applicant has included sufficient DAC/ITAAC in DCD Tier 1, Revision 9, to verify that the design of ESBWR I&C systems is completed in compliance with the applicable requirements. Therefore, the staff concludes that the NRC regulations identified in Section 7.1.1 of this report are met.

7.1.2 Software Development Activities

7.1.2.1 *Regulatory Criteria*

The staff's acceptance of the software for safety system functions is based on (1) confirmation that acceptable plans are prepared to control software development activities, (2) evidence that the plans are followed in an acceptable software life cycle, and (3) evidence that the process produces acceptable design outputs. BTP HICB-14 provides acceptance criteria for evaluating software life cycle processes for digital computer-based I&C systems.

The acceptance criteria are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1 and 21; and Appendix B to 10 CFR Part 50, Criterion III. The acceptance criteria are also based on conforming to the guidelines of RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, and RG 1.173.

7.1.2.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Appendix 7B, states that the Q-DCIS comprises the following platforms:

- RTIF-NMS
- SSLC/ESF
- ICP (VBIF, ATWS/SLC, ICS DPV Isolation Function, and HP CRD IBF)

The N-DCIS comprises the following network segments:

- GENE (DPS)
- PIP A
- PIP B
- BOP
- PCF

These platforms and network segments comprise systems of integrated software and hardware elements. Software projects are developed for the various platforms and network segments. Project software plans control the development of each platform and network segment using a software life cycle process. NEDE-33226P and NEDE-33245P are incorporated by reference into the DCD.

NEDE-33226P and NEDE-33245P are two high level documents establishing the guidelines, restrictions, requirements, program measures, and framework for creating software life cycle plans intended for the development of the digital I&C application software. These two LTRs describe the applicant's managerial, design, development, and software quality assurance processes. They also address conformance with the NRC's review acceptance criteria provided in the SRP. NEDE-33226P describes the design and development activities, while NEDE-33245P describes the software quality assurance activities during all the software life cycle phases of the digital I&C systems.

NEDO-33217 (NEDE-33217P), "Man-Machine Interface System and Human Factors Engineering Implementation Plan," is not covered in this section but is reviewed in Chapter 18 of this report. NEDE-33217P, Section 3.3.1, states that NEDE-33226P and NEDE-33245P are governing documents for the software development activities described in this LTR.

The applicant has not included any specific project plans, documentation of completed life cycle phases, or design outputs within the scope of the design certification. Instead, NEDE-33226P and NEDE-33245P provide templates for completing specific project plans, describe the documents and the review processes to be completed for each life cycle phase, and describe the design outputs that will be produced. The applicant also provides the DAC/ITAAC in DCD Tier 1, Revision 9, Section 3.2, to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14.

7.1.2.3 *Staff Evaluation*

7.1.2.3.1 Review Method for Software Development Activities

The staff reviews project-specific software for safety systems by (1) confirming that acceptable plans are developed to control software development activities, (2) verifying that the plans are followed in an acceptable software life cycle process, and (3) confirming that the process produces acceptable design outputs. As discussed in Section 7.1.1.3.1 of this report, the NRC implements the policy (SECY-92-053) of accepting the use of DAC in lieu of detailed design information in the digital I&C area during the design certification. The staff examines the software life cycle planning, implementation, and design outputs. This information can be organized as described in BTP HICB-14.

BTP HICB-14 groups the software life cycle activities into the following eight phases:

(1) Planning
(2) Requirements
(3) Design
(4) Implementation
(5) Integration
(6) Validation
(7) Installation
(8) Operations and maintenance

The following 11 different documents detail the planning effort in BTP HICB-14:

(1) Software management plan (SMP)
(2) Software development plan (SDP)
(3) Software quality assurance plan (SQAP)
(4) Software integration plan (SIntP)
(5) Software installation plan (SInstP)
(6) Software maintenance plan (SMaintP)
(7) Software training plan (STrngP)
(8) Software operations plan (SOP)
(9) Software safety plan (SSP)
(10) Software verification and validation plan (SVVP)
(11) Software configuration management plan (SCMP)

The implementation effort produces the following four types of documents in multiple life cycle phases described in BTP HICB-14:

(1) Safety analyses
(2) V&V analyses and test reports
(3) Configuration management reports
(4) Testing activities

The following nine types of documents detail the design outputs in BTP HICB-14:

(1) Software requirements specifications (SRS)
(2) Hardware and software architecture descriptions (SAD)
(3) Software design specifications (SDS)
(4) Code listings
(5) Build documents
(6) Installation configuration tables
(7) Operations manuals
(8) Maintenance manuals
(9) Training manuals

The process discussed in BTP HICB-14 is generally sequential in flow. Figure 7.1-1 is a graphic presentation of this information.

BTP HICB-14 provides the criteria for the various life cycle documents acceptable to the staff. The specific acceptance criteria for software reviews also include other applicable RGs and standards, as listed in Section 7.1.2.1 of this report.

7.1.2.3.2 General Evaluation of Software Development Activities

In NEDE-33226P and NEDE-33245P, the applicant states that its software development activities conform to BTP HICB-14. The stated purpose of the LTRs is to provide a general template for the development of project-specific plans. The template should generically present the same information as will be found in project-specific plans. The staff's review of the LTRs is to determine that they provide this information and give adequate direction for the development of the project-specific plans.

The staff verified that NEDE-33226P and NEDE-33245P contain all planning documents from BTP HICB-14. The applicant combines BTP HICB-14, SMaintP and SOP documents into the software operations and maintenance plan (SOMP). The applicant refers to the BTP HICB-14 SInstP document as the software installation plan (SIP). NEDE-33226P details the information included in the SMP, SDP, SIntP, SIP, SOMP, and STrngP. NEDE-33245P details the information included in the SVVP, SSP, SCMP, and Software Test Plan (STP).

The applicant identifies a specific life cycle (modified waterfall model). The applicant's life cycle phases map to the life cycle activities identified in BTP HICB-14 with minor deviations. The applicant combines coding, code review, and software functional test activities (module or unit testing and integration testing) into a single Implementation Phase. The applicant performs software validation testing and integrates the software into the hardware in the Test Phase. In addition, the applicant adds a retirement phase for activities related to replacement or removal of existing software products from operation.

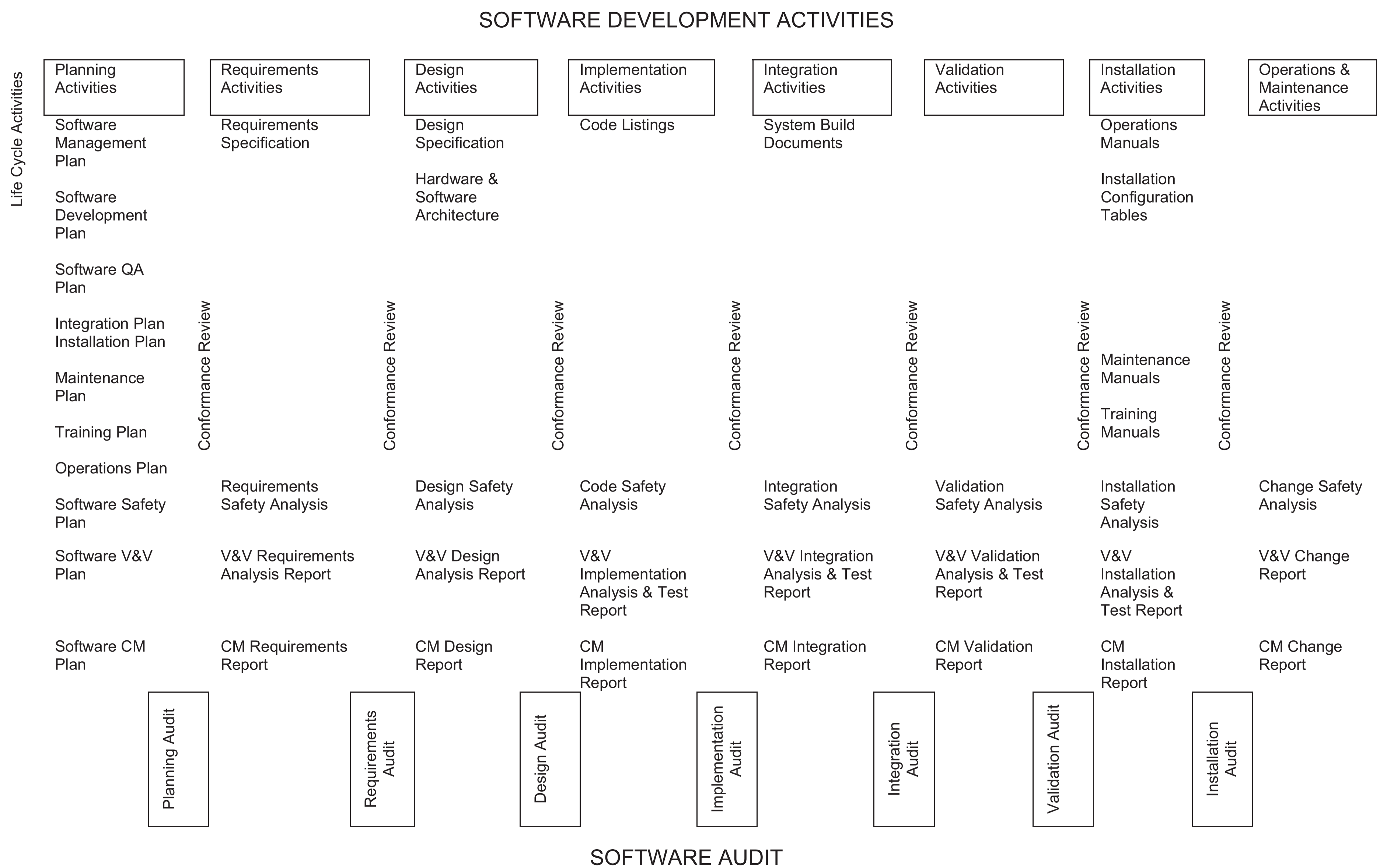

Figure 7.1-1. Example of Software Development Activities Using Generic Waterfall Life Cycle.

The applicant integrates safety analyses, V&V analysis and test reports, configuration management reports, and testing activities into the life cycle phases consistent with BTP HICB-14.

The applicant details design outputs developed in project-specific plans. The design outputs are consistent with BTP HICB-14.

The applicant utilizes engineering operating procedures and other internal, non-docketed materials as parts of NEDE-33226P and NEDE-33245P. In RAI 7.1-76, the staff requested that the applicant abstract information in the engineering operating procedures and other internal, non-docketed materials to eliminate uncertainty over unknown or unexpected internal document changes. In its response, the applicant provided abstracts for these internal procedures and policies in the topical reports, NEDE-33226P and NEDE-33245P. The staff finds that the response is acceptable since the applicant removed specific references to non-docketed material from the topical reports. Based on the applicant's response, RAI 7.1-76 is resolved. RAI 7.1-76 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that NEDE-33226P, Revision 4, and NEDE-33245P, Revision 4, included these changes and the confirmatory item is closed.

7.1.2.3.3 Evaluation of Compliance with Regulations

The staff reviewed the submitted LTRs for compliance with the requirements of GDC 21. BTP HICB-14 states that the relevant part of GDC 21 requires that protection systems be designed for high functional reliability commensurate with the safety function to be performed. The guidance in BTP HICB-14 provides an acceptable method by which the applicant can show compliance with this high functional reliability part of the requirement. Based on the applicant's commitment to follow the guidance in BTP HICB-14 as detailed in NEDE-33226P and NEDE-33245P and as confirmed by the staff in the review of these LTRs, the staff finds that the applicant has adequately addressed the requirements of GDC 21.

The staff evaluated whether 10 CFR 50.55a(1); 10 CFR 50.55a(h); GDC 1; and Appendix B to 10 CFR Part 50, Criterion III, are adequately addressed for the software development activities in accordance with BTP HICB-14. 10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.
10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995. As applied to the software development activities for 10 CFR 50.55a(a)(1) and GDC 1, BTP HICB-14 and SRP Appendix 7.1-A identify that the applicant should commit to conformance with the RGs, codes, and standards referenced in BTP HICB-14. BTP HICB-14 guidance for 10 CFR 50.55a(h) identifies the need to specify quality requirements but does not identify specific quality requirements beyond those required by 10 CFR 50.55a(a)(1) and GDC 1. As described in Section 7.1.2.3.4 of this report, the staff finds that the applicant used the applicable guidance and material provided in BTP HICB-14 to develop the material in NEDE-33245P and show compliance. GDC 1 also includes requirements for a quality assurance program and the maintenance of appropriate records. NEDE-33245P, Section 1.2, states that, for software development activities, NEDE-33245P supplements the applicant's quality assurance program. As described in Section 7.1.2.3.5 of this report, the staff finds the templates for project-specific plans acceptable; therefore, the staff finds that the applicant adequately addressed Appendix B to 10 CFR Part 50, Criterion III, for software development activities. Chapter 17 of this report addresses the staff's evaluation of the remaining portion of the quality assurance program and appropriate records. Based on the above, the staff finds that

the requirements 10 CFR 50.55a(1); 10 CFR 50.55a(h); GDC 1; and Appendix B to 10 CFR Part 50, Criterion III, are adequately addressed for the software development activities.

7.1.2.3.4 Evaluation of Deviations from Guidelines and Standards

In Appendix A, "Conformance Review," to both LTRs, the applicant documented the conformance with RGs and standards listed in BTP HICB-14 and addressed some deviations. With the exceptions of the stated deviations, NEDE-33226P and NEDE-33245P specify conformance to the criteria and guidance contained in BTP HICB-14. The applicant's stated deviations from applicable RGs and standards are summarized and evaluated below.

(1) BTP HICB-14

The applicant deviates from this guidance in the measurement implementation characteristic. The applicant states "the use of metrics to monitor development process is not fully implemented, since a proven system has not been identified."

NEDE-33226P, Section 3.6.4, "Project Controls," states, "Project Control activities include measurement and monitoring of project execution." This section also states that "project performance is monitored using computer-based tools and project reviews." Each plan template discussed in the LTRs also includes a statement that the plan will contain the definition of measurements and metrics. The staff finds that the applicant's commitment to use metrics to monitor the development process is acceptable. The particular choice of a "computer-based tool" is not an issue for the staff evaluation.

The applicant also states that the processes and activities in NEDE-33226P and NEDE-33245P are generally consistent with IEEE Std 1228, "Standard for Software Safety Plans"; IEEE Std 12207, "Standard for Information Technology - Software Life Cycle Processes"; IEEE Std 1219, "Standard for Software Maintenance"; and IEEE Std 1058, "Standard for Software Project Management Plans"; but explicit conformance is not claimed. The staff finds that the level of conformance described is acceptable.

The staff reviewed the deviations for BTP HICB-14 and finds that they are acceptable.

(2) RG 1.152

In NEDE-33226P, Revision 2, the applicant excluded IEEE Std 12207.0 and IEEE Std 603-1998 from stated conformance. The applicant describes this nonconformance as follows:

> [IEEE Std] 12207.0-1996, is not directly referenced. However, IEEE 1074-1995 is directly referenced by RG 1.173, and is therefore used instead of IEEE 12207.0. IEEE 1074-1995 covers similar topics and is the committed reference. IEEE [Std] 603-1998 addresses criteria for Safety Systems but is not within the scope of the SMP and SWAP because it does not provide guidance on software design and software quality assurance.

Since the NRC has not endorsed IEEE Std 12207-1996, and the applicant followed IEEE Std 1074, "IEEE Standard for Developing Software Life Cycle Processes," which is endorsed by RG 1.173, the staff finds this deviation is acceptable.

Conformance with IEEE Std 603-1991 is required by 10 CFR 50.55a(h). IEEE Std 603 is applicable to safety systems. Safety software is a part of the overall safety system and applies to the development process discussed in NEDE-33226P and NEDE-33245P. In RAI 7.1-78, the staff requested that the applicant directly reference IEEE Std 603-1991, and the correction sheet dated January 30, 1995 in NEDE-33226P and NEDE-33245P. In its response, the applicant stated that it will acknowledge that linkage to this standard in NEDE-33226P and NEDE-33245P. The staff finds the response is acceptable since the applicant referenced the version of the IEEE standard required by 10 CFR 50.55a(h) in NEDE-33226P and NEDE-33245P. Based on the applicant's response, RAI 7.1-78 is resolved. RAI 7.1-78 was being tracked as a confirmatory item in the SER with open items. The staff confirmed that the applicant included these changes in NEDE-33226P, Revision 4 and NEDE-33245P, Revision 4 and the confirmatory item is closed.

(3) RG 1.168

The applicant deviates from this RG in the area of software reviews. The applicant uses a different process than that outlined in IEEE Std 1028, "IEEE Standard for Software Reviews and Audits." The staff compared the applicant's review process against that of the standard and found terminology differences to be the major deviation. The necessary information for software reviews, as discussed in the standard, is found in the applicant's processes. The staff finds this deviation acceptable.

The applicant also states that "within the scope of the software plans, there is no equivalent for software integrity level 4. Detailed mapping of V&V tasks to each software classification (Q, N3, and N2) is specified in Table 2 of NEDE-33245P." NEDE-33245P discusses the applicant's Q, N3, and N2 safety classifications.

The staff's review and comparison of the tasks listed by the applicant for software classifications Q, N3, and N2 show that they are equivalent in scope to those detailed for the appropriate integrity levels of IEEE Std 1028. The staff finds this deviation acceptable.

(4) RG 1.169

The applicant deviates from the specified SCMP outline in IEEE Std 828, "IEEE Standard for Configuration Management Plans."

The staff reviewed NEDE-33245P, Section 6.0, "SCMP," against IEEE Std 828, particularly Section 5.0, "Conformance to the Standard." The general information discussed in IEEE Std 828 is found to be present in the SCMP template discussion. However, conformance cannot be claimed in accordance with IEEE Std 828, Section 5.4, "Conformance Declaration."

The main deviation is that the applicant's "sequence of information" differs from the sequence in the standard, and no explicit cross-reference is provided (IEEE Std 828, Section 5.1). Additionally, all plan information will not be included in a single document because internal applicant procedures are referenced and augmented by the SCMP.

However, the applicant's proposed outline does meet the IEEE Std 828, Section 5.3, "Consistency Criteria," and provides the basic information discussed in the standard. The deviation from Section 5.4 is considered minor. The staff finds this deviation acceptable.

(5) RG 1.172

RG 1.172 endorses IEEE Std 830-1993, "IEEE Recommended Practice for Software Requirements Specifications." Section 4.3.5 of IEEE Std 830 discusses ranking software requirements specifications for importance and stability. This ranking is used for allocation of development and design effort. The applicant does not utilize this metric to determine allocation of effort. NEDE-33245P, Section 1.5, identifies alternative criteria for the classification of software, resulting in three classes of software. In NEDE-33226P, Section 5.7.5, the applicant commits to applying an appropriate level of design and development effort to each identified requirement to "achieve a high degree of functional reliability and design quality" in accordance with IEEE Std 830. Section 7.1.2.3.7 of this report evaluates the classes of software and the level of effort applied to each, which the staff finds to be acceptable.

The staff reviewed the deviation from RG 1.172 and finds it acceptable.

(6) RG 1.173

The applicant uses names for some life cycle phases that differ from the terms in IEEE Std 1074. These differences are based on an attempt to maintain consistency with the applicant's internal procedures for software development activities, as well as address naming inconsistencies between RGs and standards. The staff's review finds that the primary differences just affect the names, and the information represented is equivalent.

The staff reviewed the deviation from RG 1.173 and finds it acceptable.

7.1.2.3.5 Evaluation of Templates for Project-Specific Plans

As stated, the primary purpose of these LTRs is to provide guidance and direction to produce project-specific plans that meet the acceptance criteria of BTP HICB-14. Therefore, the staff reviewed each document description detailed in NEDE-33226P and NEDE-33245P for expected content against the provisions of BTP HICB-14 and associated RGs. In particular, the staff examined the templates for component plans to confirm that they adequately address the management, implementation, and resource characteristics discussed in BTP HICB-14 and that NEDE-33226P and NEDE-33245P incorporated the elements of BTP HICB-14. The review of the templates for component plans also considered applicant deviations from RGs.

The staff evaluated NEDE-33226P templates for component plans as follows:

(1) SMP

Section 3.0 of NEDE-33226P discusses the SMP. The purpose of the SMP is to establish the managerial process and provide overall technical direction for software development activities. The staff reviewed the material discussed and finds that the NEDE-33226P guidance incorporates the elements of BTP HICB-14 for an SMP. The staff also finds that a project-specific SMP developed under NEDE-33226P procedures will meet the requirement for a sufficient SMP, as outlined in BTP HICB-14. Accordingly, the staff finds that the SMP component of NEDE-33226P is acceptable.

(2) SDP

Section 5.0 of NEDE-33226P discusses the SDP. The staff reviewed this section of NEDE-33226P and finds that the SDP material conforms to the guidance in IEEE Std 1074-1995, as endorsed by RG 1.173. The staff also reviewed the SDP material against RGs 1.152 and 1.172. The staff finds that a project-specific SDP developed under NEDE-33226P procedures will meet the requirement for a sufficient SDP, as outlined in BTP HICB-14. Accordingly, the staff finds that the SDP component of NEDE-33226P is acceptable.

(3) SIntP

Section 6.0 of NEDE-33226P discusses the SIntP. The purpose of the SIntP is to describe the software integration process, the hardware/software integration process, and the goals of these processes. The staff reviewed the SIntP material provided in NEDE-33226P and finds that the necessary information is discussed and identified. The staff finds that project-specific SIntPs developed under NEDE-33226P will meet the requirements for SIntPs as outlined in BTP HICB-14. Accordingly, the staff finds that the SIntP component of NEDE-33226P is acceptable.

(4) SIP

BTP HICB-14 refers to this SIP as the SInstP, which is discussed in Section 7.0 of NEDE-33226P. The staff reviewed this section of NEDE-33226P and finds that the SIP material contains the necessary information and guidance for developing project-specific SIPs, as identified in BTP HICB-14. Accordingly, the staff finds that the SIP component of NEDE-33226P is acceptable.

(5) SOMP

The SOMP is a combination of the SOP and SMaintP, as described in BTP HICB-14 and is discussed in Section 8.0 of NEDE-33226P. The activities used to operate and maintain software products during plant operation are covered by this material. The staff reviewed this section of NEDE-33226P and finds that the material contains the necessary information and guidance for developing project-specific SOMPs, as identified in BTP HICB-14. Accordingly, the staff finds that the SOMP component of NEDE-33226P is acceptable.

(6) STrngP

Section 9.0 of NEDE-33226P discusses the STrngP. The purpose of the STrngP is to ensure that adequate licensee staff training, including training for plant operators, I&C engineers, and technicians, is achieved. The staff reviewed the STrngP material provided in NEDE-33226P and finds that the necessary information is discussed and identified. The staff finds that project-specific STrngPs developed under NEDE-33226P will meet the requirements outlined in BTP HICB-14. Accordingly, the staff finds that the STrngP component of NEDE-33226P is acceptable.

The staff evaluated the NEDE-33245P templates for component plans as follows:

(7) SQAP

Section 3.0 of NEDE-33245P discusses the SQAP. The staff reviewed this section of NEDE-33245P and finds that the software quality assurance activities conform with the requirements of

Appendix B to 10 CFR Part 50 and the applicant's overall quality assurance program. Accordingly, the staff finds that the SQAP component of NEDE-33245P is acceptable.

(8) SVVP

Section 5.0 of NEDE-33245P discusses the SVVP. The staff reviewed this section of NEDE-33245P and finds that the activities are consistent with RG 1.168 and the guidance provided by IEEE Std 1012, "IEEE Standard for Software Verification and Validation." Accordingly, the staff finds that the SVVP component of NEDE-33245P is acceptable.

(9) SSP

Section 4.0 of NEDE-33245P discusses the SSP. The staff reviewed this section of NEDE-33245P and finds that the software safety activities are consistent with the guidance provided by RG 1.173 and IEEE Std 1228. Accordingly, the staff finds that the SSP component of NEDE-33245P is acceptable.

(10) SCMP

Section 6.0 of NEDE-33245P discusses the SCMP. The staff reviewed this section of NEDE-33245P and finds that the configuration management activities are consistent with the guidance provided in RG 1.169, IEEE Std 828, and IEEE Std 1042, "IEEE Guide to Software Configuration Management." Accordingly, the staff finds that the SCMP component of NEDE-33245P is acceptable.

(11) STP

BTP HICB-14 does not specifically list a separate STP. The applicant provides details on its STP in Section 7.0 of NEDE-33245P. The staff reviewed this section of NEDE-33245P and finds that the software test activities are consistent with RG 1.170, which endorses IEEE Std 829-1983, "IEEE Standard for Software Test Documentation," and RG 1.171, which endorses IEEE Std 1008-1987, "IEEE Standard for Software Unit Testing." Accordingly, the staff finds that the STP component of NEDE-33245P is acceptable.

Based on the acceptability of the guidance for the project-specific plans, NEDE-33226P and NEDE-33245P are acceptable. The staff review of the LTRs finds that these documents provide high level template guidance. This determination is based on the identification of the activities specified in BTP HICB-14 as required for project-specific plans.

7.1.2.3.6 Evaluation of Software Development Activities DAC/ITAAC

The applicant has chosen to use the DCD Tier 1 DAC/ITAAC process to allow completion of both system and project-specific design activities, as well as the completion of verifiable activities through the project-specific and system operational phases after the finalization of this evaluation.

The necessary DAC/ITAAC items are derived from the BTP HICB-14 process and resulting documents. Sufficient DAC/ITAAC are required to allow the staff to confirm that (1) acceptable plans are prepared to control software development activities, (2) the plans are followed in an acceptable software life cycle, and (3) the process produces acceptable design outputs. DCD

Tier 1, Revision 5, Section 3.2, covered these three areas, with the exception of the open item noted below.

NEDE-33226P and NEDE-33245P provide templates for completing specific project plans. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm the completion of acceptable specific project plans with the necessary management, implementation, and resource characteristics.

NEDE-33226P and NEDE-33245P describe the documents and the review processes to be completed for each life cycle phase. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to allow future confirmation that the plans are properly followed and the accomplishments documented.

NEDE-33226P and NEDE-33245P describe the design outputs that will be produced. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm that design outputs have the necessary functional and process characteristics. The DAC/ITAAC in DCD Tier 1, Revision 9, Section 3.2, also provide confirmation that the software development activities as a whole are implemented consistently with NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14.

However, in DCD Revision 5, no DAC/ITAAC items were explicitly established for creating the specific project plans from the templates. It was also not clear whether closure activities would take place on a project basis, a life cycle phase basis, or system wide. Additionally, the provided DAC/ITAAC were not clear as to the delineation between design and inspection tasks. Therefore, in RAIs 14.3-402 and 14.3-418, the staff requested the applicant to provide the DAC/ITAAC coverage for the templates and clearly relate the template to the project-specific implementation process, including the closure process. Additionally, the staff requested that the DAC/ITAAC tasks be more clearly described and criteria adequately allocated to the specific DAC or ITAAC tasks. RAI 14.3-418 was being tracked as an open item in the SER with open items. In its responses to RAIs 14.3-402 and 14.3-418, the applicant updated the DCD Tier 1 and Tier 2 documentation to align it with the software project life cycles and to ensure that each corresponding DAC will be closed before completion of a software life cycle phase. The staff finds that the responses are acceptable since the applicant aligned the DAC with the software project life cycle process described in NEDE-33226P and NEDE-33245P. Based on the applicant’s responses, RAIs 14.3-402 and 14.3-418 are resolved. The staff confirmed that the applicant included these changes in DCD Revision 6. With the resolution of the above RAIs, the staff finds the DCD Tier 1 Revision 9 DAC/ITAAC for software development activities acceptable.

7.1.2.3.7 Evaluation of Software for Nonsafety Systems

SRP Section 7.7 states that control system software should be developed using a structured process similar to that applied to safety system software. Elements of the review process may be tailored to account for the lower safety significance of control system software. NEDE-33245P, Table 1.5-1, identifies three classes of software: (1) Class Q, which is safety-related software; (2) Class N3, which is nonsafety systems software whose failure could challenge safety systems; and (3) Class N2, which is other nonsafety systems software. NEDE-33245P, Section 1.5, identifies additional criteria for the classification of software. The staff finds the software classes acceptable.

The staff verified that nonsafety systems software is developed using a structured process. NEDE-33226P and NEDE-33245P describe the software development process for all three software classes and how it varies by software class. NEDE-33245P, Tables 1-1 through 1-7, show that all software life cycle phases and design outputs identified in BTP HICB-14 are produced for the three classes.

The staff also evaluated the differences between the software classes. The primary differences between the treatment of safety and nonsafety software is in the responsibilities for performing V&V tasks and the types of V&V tasks performed. For safety software, NEDE-33245P, Section 3.2.3, states that the V&V tasks are conducted by a team that is organizationally independent of those who perform the design of the software product. For nonsafety software, NEDE-33245P, Section 3.2.3, states that the V&V tasks are conducted by individual(s) or group(s) other than those who perform the design of the software product. NEDE-33245P, Tables 1-1 through 1-7, shows how the responsibilities for performing software tasks vary by software class. NEDE-33245P, Table 2, shows how the types of V&V tasks vary by software class during the software life cycle phase. The staff finds the differences in the performance of V&V tasks based on software class acceptable.

A secondary area of the differences between software classes is in the level of qualification for development tools. NEDE-33245P, Section 4.2.9, identifies that the software tools used in the development and evaluation of software class Q and N3 software are evaluated for suitability. NEDE-33226P, Section 5.7.9, describes different levels of qualification documentation for software tools based on software class. The staff finds the differences in the qualification of software tools based on software class acceptable.

Based on the defined software classes, the implementation of the life cycle process for all software classes, and the defined differences between software classes, the staff finds that NEDE-33226P and NEDE-33245P provide a structured process for developing nonsafety system software, including control system software, that is appropriately tailored for safety significance and therefore acceptable.

7.1.2.4 *Conclusion*

The applicant has identified deviations from some of the BTP HICB-14 guidance. The staff review of these deviations finds them to be acceptable. Based on this determination, the staff finds that the applicant's DCD software development activities are consistent with BTP HICB-14 and associated regulatory guidance. Consistency with this BTP and associated guidance ensures compliance with the applicable requirements of IEEE Std 603, as required by 10 CFR 50.55a(h). As discussed in Section 7.1.2.3 above, the staff concludes that the applicant has adequately addressed the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1 and 21; and 10 CFR Part 50, Appendix B, Criterion III, for software development activities. The applicant also adequately addressed the guidelines in RG 1.152, RG 1.168, RG 1.169, RG 1.170, RG 1.171, RG 1.172, and RG 1.173.

The information provided in the NEDE-33226P and NEDE-33245P as well as the DCD Tier 1, Revision 9, DAC/ITAAC for software development activities is sufficient to confirm that (1) acceptable plans are prepared to control software development activities, (2) the plans are followed in an acceptable software life cycle, and (3) the process produces acceptable design outputs. The staff finds that there is reasonable assurance that the applicant's software development activities will result in high quality safety system software. Accordingly, the staff concludes that the software development activities and the associated DCD Tier 1, Revision 9, DAC/ITAAC are acceptable.

7.1.3 Diversity and Defense-in-Depth Assessment

7.1.3.1 *Regulatory Criteria*

In SRP Chapter 7, the staff position on D3 is established in the guidelines of BTP HICB-19. This position is based on the agency's policy prescribed in the SRM to SECY-93-087, Item II.Q. As a result of the reviews of ALWR design certification applications for designs that use a digital protection system, the NRC has established the following four-point position on D3 for digital computer-based I&C systems:

Point 1: The applicant/licensee should assess the D3 of the proposed I&C system to demonstrate that vulnerabilities to CCF are adequately addressed.

Point 2: In performing the assessment, the vendor or applicant/licensee should analyze each postulated CCF for each event that is evaluated in the accident analysis section of the SAR using best-estimate or DCD Tier 2, Chapter 15, analysis methods. The vendor or applicant/licensee should demonstrate adequate diversity within the design for each of these events.

Point 3: If a postulated CCF could disable a safety function, a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same CCF, should be required to perform either the same function as the safety system function that is vulnerable to CCF or a different function that provides adequate protection. 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.

Point 4: A set of displays and controls located in the MCR should be provided for manual system-level actuation of critical safety functions and for monitoring of parameters that support safety functions. The displays and controls should be independent and diverse from the computer-based safety systems identified in Points 1 and 3.

The purpose of BTP HICB-19 is to provide guidance for evaluating an applicant/licensee's D3 assessment and the design of manual controls and displays to ensure conformance with the NRC position on D3 for I&C systems incorporating digital computer-based RPS or ESF actuation systems. BTP HICB-19 has the objective of confirming that vulnerabilities to CCFs are addressed in accordance with the guidance of the SRM to SECY-93-087, specifically the following:

- Verify that adequate diversity is provided in a design to meet the criteria established by the NRC's requirements.

- Verify that adequate defense-in-depth is provided in a design to meet the criteria established by the NRC's requirements.

- Verify that the displays and manual controls for critical safety functions initiated by operator action are diverse from the primary protection systems.

7.1.3.2 *Summary of Technical Information*

The applicant originally submitted NEDO-33251 in July 2006. The applicant submitted subsequent revisions in August 2007, June 2009, and October 2010.

The applicant's D3 assessment is based on the following:

- PRA methods were used to consider the role of both safety and nonsafety equipment in the prevention and mitigation of transients and faults. For the design, this consideration is reflected in the overall design of the plant DCIS and mechanical systems.

- The nonsafety DPS provides a reactor trip and ESF actuations diverse from the Q-DCIS. The DPS is included to support the design risk goals by reducing the probability of a severe accident that potentially results from the unlikely coincidence of postulated transients and postulated CCFs.

NEDO-33251 provides I&C system architecture that includes the Q-DCIS and the N-DCIS. The proposed DPS is triple-redundant, nonsafety, and diverse from and independent of the Q-DCIS. The DPS provides an alternate means of initiating reactor trip and actuating selected ESF systems and providing plant information to the operator. Section 7.8 of this report addresses the staff's evaluation of the DPS.

NEDO-33251, Section 2, "Architecture/System Description," and Section 3, "Defense-in-Depth Features," which contain guidelines, requirements, and recommendations, address conformance with NUREG–0493, "A Defense-in-Depth & Diversity Assessment of the RESAR-414 Integrated Protection System," and NUREG/CR–6303, "Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems." NEDO-33251, Section 4, addresses the specific compliance with NUREG/CR–6303. NEDO-33251, Section 5, addresses specific design features to satisfy the defense-in-depth requirements. NEDO-33251, Appendix A, documents the assessment of each postulated CCF for events that are evaluated in the DCD Tier 2, Revision 9, Chapter 15, analyses assuming CCF of a digital protection system.

7.1.3.3 *Staff Evaluation*

BTP HICB-19 includes NRC's four-point position on D3 and the staff's acceptance criteria. The staff evaluated each of the four points using BTP HICB-19 as discussed below.

Point 1: The applicant/licensee should assess the D3 of the proposed I&C system to demonstrate that vulnerabilities to CCFs are adequately addressed.

The staff evaluated the applicant's D3 analysis using the criteria in BTP HICB-19, and NUREG/CR–6303. BTP HICB-19 emphasizes the review of the following topics:

(1) System Representation as Blocks

The staff evaluated the system representation as described in NEDO-33251, Sections 2.6 and 4.1 and Table 4.2. The NEDO-33251 block structure is consistent with NUREG/CR–6303, Sections 2.2 and 2.5. The staff finds this acceptable.

In NEDO-33251, Section 2.6, the applicant documents conformance to the NUREG/CR–6303 echelon of defense structure and to the NUREG/CR–6303 block structure. The four echelons are divided into three levels containing the nonsafety systems, safety systems, and nonsafety diverse systems that provide automatically and manually actuated functions to support them. The functions assigned to the I&C systems are implemented by processor-based subsystems, which are placed within a structure of separate cabinets and the DCIS rooms. The applicant maps the echelons of defense to the I&C architecture, illustrates the relationship between these subsystems and cabinets and the block structure, and shows the assignment of equipment to the blocks for each level within the echelons of defense. The staff finds that the applicant has properly mapped the echelons of defense in accordance with the NUREG/CR–6303 guidance.

(2) Documentation of Assumptions

The staff evaluated the acceptability of assumptions documented in the D3 analysis. For example, in Appendix A to NEDO-33251, the applicant provided preliminary evaluation of the DCD Tier 2, Chapter 15, AOOs and DBAs assuming CCF of the Q-DCIS. The preliminary evaluation is acceptable because the assumptions are bounded by Chapter 15 analyses. DCD Tier 1, Revision 9, Table 2.2.14-4, Item (8) documents the ITAAC for confirmatory analyses support and will validate the DPS design scope. The evaluation will be finalized when design details are final and confirmatory analyses are completed. In NEDO-33251, Section 4.7, the applicant assumes no protective action initiated as the result of CCF in the Q-DCIS, which follows the NUREG/CR–6303 approach. DCD Tier 1, Revision 5, Table 2.2.14-4, provided DAC/ITAAC for the applicant to perform FMEAs of the Q-DCIS to confirm the identified DPS functions and to perform confirmatory analyses to confirm that the DPS design ensures releases during a common mode protection system failure coincident with the DBEs discussed in DCD Chapter 15 that are within the 10 CFR Part 100, “Reactor Site Criteria,” limits (or percentage thereof), as specified in BTP HICB-19. However, it is not clear that the events and confirmatory analyses are related to specific I&C such that the DAC process would be applicable. In RAI 7.1-135, the staff requested that the applicant justify the use of the DAC process for the analyses in NEDO-33251, Table A1, including clarifying how each event is related to specific I&C. RAI 7.1-135 was being tracked as an open item in the SER with open items. In its response, the applicant clarified that the DCD Chapter 15 events were sufficiently analyzed to determine what DPS actions are required to meet the radiological criteria and that the confirmatory evaluation does not fall under the DAC process. The applicant revised DCD Tier 1, Revision 9, Table 2.2.14-4, to no longer designate any items as DAC. The separate ITAAC for confirmatory analyses and FMEAs were consolidated into an ITAAC for the applicant to complete a FMEA of the Q-DCIS, in accordance with NUREG/CR–6303 to validate the DPS protection functions. The staff finds the response is acceptable because the revised ITAAC in DCD Tier 1 Table 2.2.14-4, Item (8) covers the complete DPS design scope. Based on the applicant’s response, RAI 7.1-135 is resolved. The staff confirmed that Revision 2 of NEDO-33251 clearly identifies that the DCD Tier 2, Revision 9, Chapter 15 events are evaluated based on the credible failures identified from the final protection system design. The applicant deleted Table A1 in NEDO-33251, to eliminate the ambiguity. Based on the changes made in Revision 2 of NEDO-33251, the staff finds the applicant’s documentation of assumptions acceptable.

(3) Postulated CCFs

The staff evaluated the selection of CCFs used in the analysis. (Note that with regard to D3 analyses, the terms “CCFs” and “common mode failures (CMFs)” are used interchangeably.) NEDO-33251, Section 4, describes the following CCF scenarios considered in the NUREG/CR–6303 analysis:

- Postulated CCF of processor-based subsystems (failure occurs in all similar subsystems) - Entire system fails to perform protective functions.

- Postulated CCF in I&C architecture, in conjunction with random failures - Results are pending final hardware/software selection; however, PRA results are favorable with regard to CCFs.

- Postulated CCF within the Q-DCIS - No protective actions are initiated; the DPS provides protective actions.

- Postulated accident in conjunction with CCF of the Q-DCIS and the DPS failure - I&C strategy still enables safe shutdown (with operator input).

- Postulated event (requiring reactor trip) with CCF in the RPS function of the Q-DCIS - DPS trips reactor.

- Postulated event (requiring ESF) with CCF in the SSLC/ESF function of the Q-DCIS - DPS initiates ESF.

The staff finds the selection of CCF scenarios to be acceptable.

(4) Effect of Other Blocks

The staff evaluated the blocks assumed to function correctly. NEDO-33251, Section 4.7, states that, within the ESBWR I&C architecture, with no sharing of signals between the safety systems and the DPS, CCFs within the Q-DCIS would prevent the Q-DCIS from initiating any protective action (a conservative assumption). The staff finds the treatment of other blocks acceptable.

(5) Identification of Alternate Trip or Initiation Sequences

The staff evaluated the selection of sequences in NEDO-33251, Appendix A. The event sequences evaluated are consistent with DCD Tier 2, Revision 4, Chapter 15. DCD Tier 2, Revision 5, Chapter 15, reordered its list of sequences. For example, DCD Tier 2, Revision 4, Chapter 15, and NEDO-33251, Appendices A and B, identify no reactor and power distribution anomalies. However, DCD Tier 2, Revision 5, Chapter 15, identifies two reactor and power distribution anomalies, "Control Rod Withdrawal Error During Startup," and "Control Rod Withdrawal Error During Power Operation." In RAI 7.1-131, the staff requested that the applicant revise NEDO-33251 to ensure that the events and accidents evaluated in the D3 analysis are consistent with DCD Tier 2, Chapter 15. The staff has not identified the need for additional sequences. RAI 7.1-131 was being tracked as an open item in the SER with open items. In its response, the applicant indicated that it would update the evaluation of Chapter 15 events in NEDO-33251 to be consistent with DCD Tier 2, Revision 9, Chapter 15. The staff finds that the response is acceptable since the applicant revised NEDO-33251 to be consistent with DCD Tier 2, Revision Chapter 15. Based on the applicant's response, RAI 7.1-131 is resolved. Based on the changes made in Revision 2 of NEDO-33251, the staff finds the applicant's identification of alternate trip or initiation sequences acceptable.

(6) Identification of Alternative Mitigation Capability

The staff evaluated the selection of alternative mitigation actuation functions. NEDO-33251, Appendix A, describes the potential CCF for each DBE in DCD Tier 2, Revision 9, Chapter 15.

Appendix A also describes the associated alternative mitigation function provided by the DPS to prevent or mitigate core damage and unacceptable release of radioactivity. DCD Tier 2, Revision 9, Section 7.8.1.2, describes comparable alternative mitigation functions in the description of the DPS. NEDO-33251, Appendix A, identifies that the D3 analysis needs to be updated when design details are finalized (e.g., hardware platforms and details of the hardware components are determined, and failure modes and effects are better known or evaluated). DCD Tier 1, Revision 9, Table 2.2.14-4 includes the ITAAC for the applicant to perform FMEAs of the Q-DCIS to confirm the identified DPS functions. The staff finds the identification of alternative mitigation capability acceptable.

In conclusion, the staff finds that the applicant appropriately addressed the review topics emphasized in BTP HICB-19 and the guidelines of NUREG/CR–6303. Accordingly, the staff finds that Point 1 is adequately addressed.

Point 2: In performing the assessment, the vendor or applicant/licensee should analyze each postulated CCFs for each event that is evaluated in the accident analysis section of the SAR using best-estimate or SAR Chapter 15 analysis methods. The vendor or applicant/licensee should demonstrate adequate diversity within the design for each of these events.

The staff evaluated the methods used to analyze postulated failures. NEDO-33251 uses best-estimate methods. The staff finds the use of these methods acceptable since it is consistent with the NRC position. The staff evaluated the applicant's demonstration of diversity. The applicant uses a three-layered diversity approach as outlined in NEDO-33251, Section 5.2.1: (1) the N-DCIS for monitoring and control of nonsafety functions; (2) the Q-DCIS for reactor trip, ESF, and safety monitoring; and (3) the DPS for nonsafety reactor trip functions, actuation of ESF, and operator displays. The DPS is specifically implemented in hardware and software that is diverse from and independent of that used in the Q-DCIS. The staff finds the demonstration of diversity acceptable. Accordingly, the staff finds that Point 2 is adequately addressed.

Point 3: If a postulated CCF could disable a safety function, a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same CCF, should be required to perform either the same function as the safety system function that is vulnerable to CCFs or a different function that provides adequate protection. 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 staff evaluated the means to ensure sufficient quality to perform the necessary function under the associated event conditions. As an example, NEDO-33251, Section 5.3, and DCD Tier 2, Revision 9, Section 7.8.1, identify the use of the ATWS/SLC logic to accomplish the reactor shutdown function as a diverse method from the Q-DCIS platform (the RPS shutdown). The staff finds this acceptable. The quality of the ATWS/SLC logic is evaluated in Section 7.8 of this report. Accordingly, the staff finds that Point 3 is adequately addressed.

Point 4: A set of displays and controls located in the MCR should be provided for manual system-level actuation of critical safety functions and for monitoring of parameters that support safety functions. The displays and controls should be independent and diverse from the computer-based safety systems identified in Points 1 and 3.

The staff evaluated whether a set of displays and controls located in the MCR is provided for manual system-level actuation of critical safety functions and for monitoring of parameters that support safety functions. DCD Tier 2, Revision 9, Section 7.8.1, identifies that the DPS provides diverse monitoring and indication of critical safety functions and process parameters required to support manual operations and assessment of plant status. Additionally, all safety systems have displays and controls located in the MCR that provide manual system-level actuation of their safety functions and monitoring of parameters that support those safety functions. The staff finds this to be acceptable.

The staff evaluated whether the displays and controls are diverse from and independent of the computer-based safety systems identified in Points 1 and 3.

In addition to the manual controls and displays for the safety reactor trip and ESF functions, DCD Tier 2, Revision 9, Section 7.8.1, states that the DPS also has displays and manual control functions that are diverse from and independent of those of the safety protection and SSLC/ESF functions. They are not subject to the same CCF as the safety protection system components. The manual controls include the manual initiation of the SRV, DPV, GDCS, and SLC system valves, and the ICS. The operator is provided with a set of diverse displays separate from those supplied through the safety platforms. The DPS displays provide independent confirmation of the status of major process parameters. The staff finds this acceptable. Accordingly, the staff finds that Point 4 is adequately addressed.

7.1.3.4 *Conclusion*

Based on the review of the DCD and NEDO-33251, the staff finds that the applicant has adequately addressed the relevant guidelines of SRM to SECY-93-087, Item II.Q, and BTP HICB-19 (including the NRC's D3 four-point position). The applicant addressed how the design conforms to the guidelines and recommendations discussed in NUREG/CR–6303. Therefore, the staff finds the applicant's D3 assessment acceptable.

7.1.4 Setpoint Methodology

7.1.4.1 *Regulatory Criteria*

The objective of the review of NEDE-33304P, "ESBWR Setpoint Methodology," Revision 4 is to confirm that the applicant's setpoint methodology satisfies regulatory acceptance criteria, guidelines, and performance requirements to protect the integrity of physical barriers that guard against the uncontrolled release of radioactivity. The staff evaluated the applicant's setpoint methodology based on the guidelines prescribed in BTP HICB-12. The following regulatory requirements and guidance documents apply to the staff's review of the applicant's ESBWR setpoint methodology:

- GDC 13 requires, in part, that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

- GDC 20 requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs.

As required by 10 CFR 50.36(c)(1)(ii)(A), the TS must include limiting safety systems settings (LSSSs). This paragraph specifies, in part, that, "where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Accordingly, the setpoint methodology must properly establish the setpoints for instrument channels that initiate protective functions.

As stated in 10 CFR 50.36(c)(3), the surveillance requirements "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

In addition, 10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995. Section 4.4 of IEEE Std 603 requires identification of the analytical limit associated with each variable. Section 6.8.1 requires that allowances for uncertainties between the analytical limit and device setpoint be determined using a documented methodology.

RG 1.105 describes a method acceptable to the staff for complying with the NRC's regulations for ensuring that setpoints for safety instrumentation are initially within and remain within the TS. This RG endorses Part 1 of ISA-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants." ISA-S67.04-1994, Part 2, "Methodology for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," provides additional guidance, but is not endorsed by the staff.

7.1.4.2 *Summary of Technical Information*

The applicant's setpoint methodology computation method is based on a statistical, probabilistic approach. The setpoint methodology combines the uncertainty components to determine the allowance and trip settings including tolerances for the functions of the safety-related systems. All appropriate and applicable uncertainties are considered for each safety-related 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. The basic algorithm to combine the independent and random uncertainty components is the square-root of the sum of the squares (SRSS) technique. The uncertainties that are not independent are conservatively treated by arithmetic summation and then systematically combined with the independent terms. The appropriate uncertainties, as defined by a review of the plant baseline design input documentation, are included in each safety function uncertainty calculation. This setpoint methodology utilized ISA-RP67.04.02-2000, "Setpoint for Nuclear Safety-Related Instrumentation" as a guideline. The latest version of RG 1.105, Revision 3, endorses Part 1 of ISA-S67.04-1994.

The applicant's setpoint methodology describes the establishment of setpoints and the relationships between nominal trip setpoints (NTSPs), allowable value (AV), as-left tolerance (ALT), as-found tolerance (AFT), analytical limit (AL), and safety limit.

The safety limits are chosen to protect the integrity of physical barriers that guard against the uncontrolled release of radioactivity. The safety limits are typically provided in the plant safety analyses. The analytical limit is established to ensure that the safety limit is not exceeded. The analytical limit is developed from event analyses models that consider parameters including process delays, rod insertion times, reactivity changes, and instrument response times.

The purpose of the AV is satisfied by providing enough allowance for the AL to account for those uncertainties not measured during periodic testing (channel operational test, channel functional test, and calibration test) to protect the safety limit. The AV is derived from the AL by subtracting (or adding) the SRSS of all instrument errors except drift that are not measured during periodic testing. The AV is the value at which the instrument channel should be evaluated for operability to protect the safety limit when the test is performed. These periodic surveillance tests provide assurance that the analytical limit will not be exceeded if the AV is satisfied.

The limiting trip setpoint (LTSP) is the equal to the first nominal trip setpoint ($NTSP_1$) and is the final nominal trip setpoint ($NTSP_F$) value with the minimum required allowance to AL. The second nominal trip setpoint ($NTSP_2$) is derived from the AV by subtracting the maximum allowance AFT (AFTmax) which is defined by the SRSS of channel instrument accuracy, measurement and test equipment accuracy including error and readability, and channel instrument drift. The allowance, designated as the AFT, between the AV and $NTSP_F$ is sufficient to assure that the AV is not exceeded during surveillance testing and is small enough not to mask channel degradation. The $NTSP_F$ is derived from the AV by subtracting (adding) the AFT. The $NTSP_F$ must be more conservative than the LTSP and is between $NTSP_1$ and $NTSP_2$. The AFT is derived from assumption or design inputs used in the trip setpoint calculations that are intended to assure a high level of confidence in future acceptable channel performance. The ALT is established by the necessary accuracy band (calibration accuracy); therefore, a device or instrument channel must be calibrated within the $NTSP_F$ during surveillance. The maximum allowance for ALT is the SRSS of channel instrument accuracy and measurement and test equipment accuracy. The as-left condition is the condition in which the instrument channel is left after calibration or trip setpoint verification. Additionally, if the as-found value is within the as-left tolerance, then recalibration is not required. The channel will be considered inoperable if the as-found value is outside the AFT.

As stated in 10 CFR 50.36(c)(1)(ii)(A), the LSSSs are settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which a safety limit is placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before the safety limit is exceeded. In the applicant's methodology, the $NTSP_F$ is established to ensure that an instrument channel trip signal occurs before the safety limit is reached and to minimize spurious trips close to the normal operating point of the process.

7.1.4.3 *Staff Evaluation*

The ESBWR setpoint methodology provides acceptable criteria as follows: (1) the $NTSP_F$ must be between $NTSP_1$ and $NTSP_2$; (2) the $NTSP_F$ is the AV minus the AFT; and (3) the AFT is equal to or larger than the ALT. The staff evaluates if the applicant's setpoint methodology provides the $NTSP_F$ as the LSSS and the AV to comply with 10 CFR 50.36(c)(1)(ii)(A). The setpoint methodology provides the $NTSP_F$, AV, AFT, and ALT for the surveillance to comply with the requirement of 10 CFR 50.36(c)(3). The setpoint methodology provides the $NTSP_F$ to comply with the requirements of GDC 13 and 20. The setpoint methodology provides the trip setpoints to comply with the requirements stated in Sections 4.4 and 6.8.1 of IEEE Std 603. All appropriate and applicable uncertainties are considered for each safety 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.

NEDE-33304P, Revision 0, identified that a graded approach would be used to apply different levels of technical rigor, probability, and confidence to various setpoints. In RAI 7.1-86, the staff requested that the applicant verify that the setpoint methodology can establish setpoints with the 95/95 tolerance limit, consistent with RG 1.105 for uncertainties for each of the categories in the graded approach. In the RAI response, the applicant revised its setpoint methodology to address only the scope of all safety automatic protective device settings as well as all automatic protective device settings that meet the requirements of 10 CFR 50.36(c)(1)(ii) for TS required limiting safety system settings.

The applicant takes a conservative approach to establishing its setpoint methodology by adding the calibration accuracy to an allowance between the analytical limit and AV. However, the setpoint methodology uses one-sided normally distributed probability at the 95 percent level, which will have 95 percent of the uncertainties falling between negative infinity and +1.645 standard deviations in the development of nominal trip setpoints and AVs. The allowances (margins) for the setpoint calculation, using one-sided, normally distributed probability (as compared to two-sided) are decreased from 0.98 (1.960/2) to 0.82 (1.645/2). The applicant's setpoint methodology is based on General Electric's Instrument Setpoint Methodology, described in NEDC-31336P-A, issued September 1996, which utilizes single-sided distributions in the development of trip setpoints and AVs. The staff SER states that NEDC-31336P-A is acceptable provided that a channel approaches a trip in one direction. In RAI 7.1-102, the staff requested that the applicant provide a clear and detailed justification for the application of a one-sided distribution to its setpoint methodology.

The staff found that the applicant had not demonstrated that the ESBWR setpoint methodology conforms to the 95/95 tolerance limit as an acceptable criterion for uncertainties specified in RG 1.105, Revision 3. To provide an independent evaluation, the staff contracted Oak Ridge National Laboratory (ORNL) to develop a technical evaluation report (TER) evaluating the applicant's methodology using the guidance in RG 1.105, Revision 3. The ORNL report, "TER for GE-Hitachi's Setpoint Methodology NEDE-33304P," confirmed the staff's findings; therefore, the staff requested, in RAI 7.1-141, that the applicant revise the setpoint methodology to remove the reduction factor of 1.645/2 or provide an alternative to the RG 1.105, Revision 3, acceptance criteria. RAI 7.1-141, which included the ORNL report as an attachment, superseded and closed RAIs 7.1-86 and 7.1-102. In response to RAI 7.1-141, the applicant revised NEDE-33304P to remove the reduction factor of 1.645/2 and to make corresponding changes to the supporting information. The staff finds the response is acceptable since the applicant revised its setpoint methodology to conform to the guidelines of RG 1.105, Revision 3. Based on the applicant's responses, RAI 7.1-86, 7.1-102, and 7.1-141 are resolved.

DCD Tier 1, Revision 9, Table 2.2.15-2, Item 21, includes the DAC/ITAAC for verifying compliance with IEEE Std 603, Section 6.8.

7.1.4.4 *Conclusion*

Based on the review of information in NEDE-33304P, "GEH ESBWR Setpoint Methodology," Revision 4, the staff concludes that the ESBWR setpoint methodology is acceptable and meets the applicable regulatory requirements of 10 CFR 50.36, 10 CFR 50.55a(h), and GDC 13 and 20. The staff also finds that ESBWR setpoint methodology establishes the trip setpoint so that automatic protective action will correct the abnormal situation to protect the safety limit.

7.1.5 Data Communication Systems

The data communication functions are embedded within the Q-DCIS and the N-DCIS architecture. Sections 7.1, 7.2, and 7.3 of DCD Tier 2, Revision 9, discuss many of the data communication functions. The staff used SRP Section 7.9, to review the acceptability of the data communication functions of the DCIS.

7.1.5.1 *Regulatory Criteria*

The objectives of the review are to confirm that the DCIS meets the following criteria:

- Conform to applicable acceptance criteria and guidelines
- Perform the safety functions assigned to them
- Meet the reliability and availability goals assumed for the system
- Tolerate the effects of random transmission failures

SRP Table 7-1 identifies the regulatory requirements. It states that the data communication systems addressed by SRP Section 7.9 are support systems for one or more of the systems addressed by SRP Sections 7.2 through 7.8. Acceptance criteria for a specific data communication system are derived from the acceptance criteria for the systems supported by that data communication system.

The acceptance criteria are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(v); 10 CFR 50.62; GDC 1, 2, 4, 13, 15, 19, 21, 22, 23, 24, 28, and 29; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152, and the SRM to SECY-93-087, Item II.Q.

A potential concern is that the transmission of multiple signals over a single path may constitute a single point of failure that could have a larger impact on plant safety than would occur in analog systems.

7.1.5.2 *Summary of Technical Information*

The Q-DCIS provides the data processing and transmission network that encompasses the four independent and separate data multiplexing divisions, Divisions 1, 2, 3, and 4, corresponding to the four divisions of safety electrical and I&C equipment. Each Q-DCIS division consists of the RMUs, the fiber optic cable signal transmission path, the SSLC/ESF cabinets, the RTIF cabinets, NMS cabinets, the cabinet power supplies, the safety VDUs, and safety fiber optic CIMs.

The Q-DCIS contains multiple dual-redundant fiber optic cable networks for each of the four divisions. The networks connect the RMUs with the divisional safety VDUs, the digital trip modules (DTMs), the CIMs, the safety logic test cabinets, and the N-DCIS through isolated digital gateways and datalinks.

Each Q-DCIS system is housed in a set of uniquely identified cabinets. Separate cabinets are provided for each of the four divisions and the remotely mounted components within each division.

The field sensors and process transmitters are hard-wired to the divisional local RMUs in the reactor and control buildings. At the input module of the field RMUs, the analog data are delivered to the analog input modules, and discrete data are delivered to the digital input modules. The field sensors and wiring belong to the process system to which they are attached and are not part of the Q-DCIS. Analog signal conditioning, analog-to-digital conversion, and digital signal conditioning such as filtering and voltage level conversion are performed at the input modules.

Each field RMU formats and transmits input signals as data messages to the dual network and then to the RTIF, NMS, SSLC/ESF, and ICP components within its own division. The field RMUs receive the SSLC/ESF equipment control signals from the network for distribution by hard-wired connection to the equipment actuators of the ESF functions.

The corresponding divisional Q-DCIS networks send data to the RTIF, NMS, and SSLC/ESF components in separate RTIF, NMS, and SSLC/ESF divisional cabinets. The data are also sent to other safety logic equipment such as the safety logic test cabinets, for control of the functional tests, the CIMs, and the isolated divisional gateways and datalinks for communication with the N-DCIS.

The N-DCIS includes a nonsafety network that is segmented into parts that can work independently of one another if failures occur. The segments are not visible to the operator during normal operation. The N-DCIS uses hardware and software platforms that differ from the Q-DCIS. The N-DCIS network is dual redundant and redundantly powered. The following are the individual N-DCIS segments:

- GENE network
- PIP A network
- PIP B network
- BOP network
- Plant computer network

The segments are redundant, managed network switches into which the data acquisition, control, and displays associated with that segment are connected. All connections to these switches are through the fiber optic cable network. The switches allow the various controllers, data acquisition, and displays associated with a segment to communicate with each other. The switches' "backbone" capacity determines how many simultaneous two-way connections can be made. Only when a switch determines that an information data packet is destined for a node on another switch is the information put on an uplink to that switch. The network switches learn and maintain their own forwarding tables containing a list of all the nodes and hosts on their respective network segment. When a network switch receives a data communication packet, it forwards only that particular data communication packet to the segment to which that receiving host is connected. This mechanism prevents data traffic between devices on the network from impacting devices on other segments of the network. Specifically, the switches use a "spanning tree protocol" to automatically enable and disable ports so there is normally only one path from the nodes of one switch to another. Should a path become disabled, the switches automatically reconfigure to establish another path through the remaining switches and fiber optic cable paths. Reconfiguration requires no operator input.

The N-DCIS is not a single network. It is redundant and segmented to support the DCIS. A single failure of one of the redundant switches in a segment or multiple failures that involve no more than one switch per segment have no effect on plant operation or data. The failure is alarmed and can be repaired online. If both switches of a segment simultaneously fail, that particular segment is lost. However, the remaining segments are unaffected, and individual nodes connected to the failed switches may continue to function. The remaining switches then automatically reconfigure their uplink ports such that the remaining segments automatically find data communication paths between themselves.

7.1.5.3 *Staff Evaluation*

7.1.5.3.1 Evaluation of Data Communication System Conformance with Acceptance Criteria - Major Design Considerations

SRP Section 7.9 lists the following 13 major design considerations that should be emphasized in the review.

(1) Quality of Components and Modules (IEEE Std 603, Section 5.3)

Section 7.1.1.3.10 of this report discusses the staff's evaluation of the quality of components and modules presented by the quality assurance program with regard to evaluation of conformance to IEEE Std 603, Section 5.3. The applicant also stated that the quality assurance program conforms to GDC 1. Chapter 17 of this report addresses the evaluation of the adequacy of the quality assurance program. These evaluations apply to the DCIS. Accordingly, the staff finds that the quality of the components and modules design consideration is adequately addressed.

(2) DCIS Software Quality (IEEE Std 7-4.3.2, Section 5.3)

Section 7.1.1.3.10 of this report discusses the staff's evaluation of software quality with regard to evaluation of conformance to IEEE Std 603, Section 5.3 and IEEE Std 7-4.3.2, Section 5.3. This evaluation applies to the DCIS. Accordingly, based on the applicant's use of an acceptable software development process, as evaluated in Section 7.1.2 of this report, and its verification in the DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC, the staff finds that the DCIS software quality design consideration is adequately addressed.

(3) Performance (IEEE Std 603, Section 5.5)

The staff evaluated whether issues related to real-time performance are adequately addressed for the DCIS data communication systems. Section 7.1.1.3.10 of this report discusses the staff's evaluation of software quality with regard to evaluation of conformance to IEEE Std 603, Section 5.5. This evaluation applies to the DCIS data communication systems. The real-time performance should be reviewed with BTP HICB-21. DCD Tier 2, Revision 9, Section 7.1.6.5, states that the system conforms to BTP HICB-21. BTP HICB-21 notes that (1) time delays within the DCIS and measurement inaccuracies introduced by the DCIS should be considered when reviewing setpoints, (2) data rates and data bandwidths should be reviewed, including impact from environmental extremes, and (3) sufficient excess capacity margins should be available to accommodate future increases. The applicant provided assurances that the safety networks will be completely deterministic (i.e., time based as opposed to event based), as indicated in DCD Tier 2, Revision 9, Section 7.1.3.2.7. This section states that the Q-DCIS internal and external communication protocols are deterministic (i.e., time based as opposed to event based), which is consistent with BTP HICB-21 guidelines.

In RAI 7.9-10, the staff asked the applicant to provide the design guidelines and the design approach concerning sufficient spare memory and speed (of the processors). In its response, the applicant stated that NEDE-33226P and NEDE-33245P define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Table 3.2-1, provides for verification of activities associated with NEDE-33226P and NEDE-33245P. The response also states the following:

> For non deterministic links, which may exist as part of the N-DCIS, the networks and switches will be tested in an environment that includes large amounts of extraneous data to verify that no information needed for plant safety or control is lost. The use of managed network switches in the N-DCIS networks (as indicated in DCD Tier 2, Subsection 7.1.5.2) prevents excessive or unexpected data on these networks.

The staff finds the response is acceptable since the applicant identified an approach to ensure sufficient spare memory and speed of the processors. Based on the applicant's response, RAI 7.9-10 is resolved.

Accordingly, based on the applicant's use of an acceptable software development process, as evaluated in Section 7.1.2.3 of this report, and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the performance design consideration is adequately addressed.

(4) Reliability (IEEE Std 603, Section 5.15)

The staff evaluated whether the reliability design consideration is adequately addressed for the DCIS. In accordance with SRP Section 7.9, the staff reviewed the effects of unneeded functions; effects of error detection and recovery; and how corrupted, missing, or duplicate messages are detected and repaired. In addition, the operating history of the DCIS in similar applications should be determined to be satisfactory, but DCD Tier 2, Revision 9, did not address this issue. Instead, the applicant states that its commitment to reliability is ensured by the functional reliability and equipment reliability provided under the applicant's 10 CFR Part 50, Appendix B, quality assurance program. The applicant also states that following BTP HICB-14 guidance for software development, processes will achieve reliable software design and implementation. Section 7.1.1.3.10 of this report discusses the staff's evaluation of the reliability design consideration with regard to evaluation of conformance to IEEE Std 603, Section 5.15. This evaluation applies to the data communication systems. Accordingly, the staff finds that the reliability design consideration is adequately addressed.

(5) Time Coherency of Data

As described in NEDE-33226P, communication protocols are developed as part of the software life cycle process, and DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying the implementation of the software life cycle process, including products such as communication protocols. Therefore, the applicant's methods to ensure the correct sequence of data packets at receiving data communication nodes can be verified by the ITAAC process. Accordingly, the staff finds that the time coherency of data design consideration is adequately addressed.

(6) Control of Access (IEEE Std 603, Section 5.9)

The staff evaluated whether control of access is adequately addressed for the DCIS communication systems. Section 7.1.1.3.10 of this report evaluates Section 5.9 of IEEE Std 603. This evaluation applies to the data communication systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.10, does provide general assurances for physical access control of the DCIS. Administrative control is used to implement access control to vital areas of the plant, including the MCR. Physical security and electronic security devices are provided to ensure that only authorized and qualified plant personnel are allowed to have access to the Q-DCIS cabinets and consoles. The Q-DCIS equipment has its own access control devices. The Q-DCIS cabinets have doors with key locks and position switches. The Q-DCIS components within the cabinets have key lock switches that are used to control access to special functions.

Keys, passwords, and other security devices (following the guidance of RG 1.152) are used to control access to specific rooms; open specific equipment cabinets; obtain permission for access to enter specific electronic instruments for calibration, testing, and setpoint changes; and gain access to safety system software and data. Safety software is not routinely changed at the plant site. Opening a Q-DCIS cabinet door produces an alarm in the MCR. No access to safety system equipment and control can be gained through the network from nonsafety system equipment. Computer-related access controls and authorization are part of the cyber-security program plan, which is described in NEDO-33295 and NEDE-33295P. In RAI 7.1-80, the staff asked the applicant to specifically verify that there will be no remote access to any safety systems. RAI 7.1-80 was being tracked as an open item in the SER with open items. In its response, the applicant added a statement to NEDE-33295P to clarify that that there will be no remote access to safety systems. The staff finds the response is acceptable since the applicant modified NEDE-33295P to clarify that that there will be no remote access to safety systems. Based on the applicant’s response, RAI 7.1-80 is resolved. The staff confirmed that the applicant included these changes in NEDE-33295P, Revision 1. Based on the above, the staff finds that the control of access consideration is adequately addressed.

(7) Single Failure Criterion (IEEE Std 603, Section 5.1)

The staff evaluated whether the single failure criterion, IEEE Std 603, Section 5.1, is adequately addressed for the DCIS communication systems. The Q-DCIS contains dual redundant data communication channels per division and four redundant divisions. With the commitment to this conformity, the staff believes that the channel assignments to individual communication subsystems are appropriate to assure that requirements for redundancy and diversity are met. Section 7.1.1.3.10 of this report discusses the staff’s evaluation of the single failure criterion with regard to evaluation of conformance to IEEE Std 603, Section 5.1. This evaluation applies to the data communication systems. Accordingly, the staff finds that the single failure criterion consideration is adequately addressed.

(8) Independence (IEEE Std 603, Section 5.6)

The staff evaluated whether the independence criterion, IEEE Std 603, Section 5.6, is adequately addressed for the DCIS communication systems. This criterion is evaluated in Section 7.1.1.3.10 of this report, where the staff finds that Section 5.6 is adequately addressed on the basis of its inclusion in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC. This evaluation applies to the data communication systems. In addition, the staff evaluated whether nonsafety data communications could interfere with safety data communications consistent with NUREG/CR–6082, “Data Communications,” Section 2.1.9.1, which states, “Interference can occur even if supposedly independent data communications systems exchange only ‘handshakes’ or synchronizing signals.” DCD Tier 2, Revision 9, Section 7.1.3.3.4, identifies time tagging as one of the two types of signals sent from nonsafety to safety components in the DCIS. (Nonsafety calibration data are sent from the 3D Monicore function to the safety NMS; however, the information exchange is manual and rigorously controlled.) DCD Tier 2, Revision 9, Section 7.1.3.3, states the following:

> Time signals are sent to the Q-DCIS safety fiber optic CIMs through the non-safety gateways for display on the Q-DCIS (SSLC/ESF) safety VDUs and for use by the Q-DCIS to allow time tagging of data sent to the N-DCIS. These time signals are only used by the Q-DCIS for VDU indication so that all displays show the same time of day. The time signals sent from the N-DCIS to the Q-DCIS are never used to synchronize logic nor is the safety logic dependent in any way on the absence, presence, or correctness of the time signal.

In RAI 7.9-8 and RAI 7.9-8 S01, the staff requested the applicant to clarify its approach to time tagging. In its responses, the applicant stated the following:

> “The design process described in NEDE-33226P and NEDE-33245P provides hardware/software development, construction, testing, and approval processes that ensure that any time tagging delays are within the design requirements specified for each system.”

The staff finds the response is acceptable since the applicant clarified its approach to time tagging. Based on the applicant’s response, RAI 7.9-8 is resolved. DCD Tier 1, Revision 9, Table 3.2-1, provides for verification of activities associated with NEDE-33226P and NEDE-33245P. Accordingly, the staff finds that the independence design consideration is adequately addressed.

(9) System Testing and Inoperable Surveillance (IEEE Std 603, Sections 5.7, 5.8, and 6.5)

The staff evaluated whether the system testing and inoperable surveillance design consideration is adequately addressed for the DCIS communication systems. In accordance with SRP Section 7.9, the system testing and inoperable surveillance design consideration is addressed by conformance with IEEE Std 603, Sections 5.7, 5.8 and 6.5. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, Sections 5.7, 5.8 and 6.5. This evaluation applies to the data communication systems. Accordingly, the staff finds that the system testing and inoperable surveillance design consideration is adequately addressed.

(10) Protocols

The staff evaluated whether the protocols design consideration is adequately addressed in the software development activities. DCD Tier 2, Revision 9, Section 7.1.3.2.7, states that the Q-DCIS internal and external communication protocols are deterministic. In response to RAI 7.9-10, the applicant stated, “although the N-DCIS is not deterministic, the 100 megabit Ethernet ports and dedicated RMUs to control processor communications make the design almost so.” The response goes on to justify this statement. In NEDE-33226P, Sections 5.7.7 and 5.8.3.2 specify the characteristics of data communication protocols and intrasystem communication protocols. NEDE-33226P specifies that the external data communication protocol specification is a requirements phase output document and the intrasystem communication protocols specification is a design phase output document. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm the implementation of software development activities. Accordingly, the staff finds that the protocols design consideration is adequately addressed.

(11) EMI/Radiofrequency Interference (RFI) Susceptibility (IEEE Std 603, Section 5.4)

The staff evaluated whether the EMI/RFI susceptibility criterion is adequately addressed for the DCIS communication systems. This criterion is part of EQ and Section 5.4 of IEEE Std 603. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, Section 5.4. In addition, the staff evaluated whether the hardware/software specification includes the specification that fiber-optic-related materials do not become brittle under radiation. NEDE 33226P, in its description of the SDP, identifies that the hardware/software specification, which includes cabling requirements, will be a requirements phase output document. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm the implementation of the SDP. Accordingly, the staff finds that the EMI/RFI susceptibility design consideration is adequately addressed.

(12) D3 Analysis

SRP Section 7.8 states that the D3 assessment and conformance to the SRM to SECY-93-087, Item II.Q, should be reviewed for data communication systems that are part of protection and diverse systems. The staff evaluation of the D3 assessment and conformance to the SRM to SECY-93-087, Item II.Q, in Sections 7.1.3 and 7.1.1.3.7 of this report applies to the data communication system. The staff finds that the D3 analysis of the data communication design is adequately addressed.

(13) DCIS Exposed to Seismic Hazard

DCD Tier 2 does not specify the location of the DCIS equipment. The staff evaluated whether the DCIS components are in seismic Category I structures. DCD Tier 2, Revision 9, Table 3.2-1, indicates that the Q-DCIS electrical modules and cables with safety functions are seismic Category I. DCD Tier 2, Table 3.2-1, also indicates that the N-DCIS components whose failure can potentially adversely affect seismic Category I components (e.g., in the MCR) are required to be seismic Category II. Accordingly, the staff finds the exposure to seismic hazard is adequately addressed.

7.1.5.3.2 Evaluation of Data Communication Systems Conformance with Acceptance Criteria - Other Criteria

SRP Section 7.9 states that the data communication system design should be evaluated for conformance to IEEE Std 603. Since Section 7.1.1.3.10 of this report provides a general evaluation of conformance to IEEE Std 603, this section focuses on the specific conformance of the data communication systems and provides additional evaluations of IEEE Std 603 criteria previously considered in this report.

The staff evaluated whether the DCIS data communication systems provide proper data isolation. In RAI 7.1-65, the staff asked the applicant to describe the CIM safety-related functions and how they will be confirmed. RAI 7.1-65 was being tracked as an open item in the SER with open items. In its response, the applicant clarified that CIMs are safety signal isolation devices. The applicant revised DCD Tier 2, Section 7.1.3.3, to clarify that the safety fiber optic CIMs are the isolation devices, including data isolation, and convert signals between electricity and light on the safety side of the fiber optic cable. These safety fiber optic CIMs are powered by the division within which they are physically located. The safety fiber optic CIMs are qualified as safety components. The applicant also revised DCD Tier 1, Revision 9, Table 2.2.15-2 Item 10, to provide DAC/ITAAC to verify that the software project's interdivisional communication systems have optically isolated fiber optical communication pathways. The staff finds the responses are acceptable since the applicant clarified the CIM safety-related functions and how they will be confirmed. Based on the applicant's responses, RAI 7.1-65 is resolved. The staff finds that the data communication design has proper data isolation provisions.

The staff evaluated whether the data communication system design has deterministic character. The deterministic character is an instrumentation response that is predictable and repeatable from sensor input to output command to the control device to actuate. For digital systems and software, a deterministic character means that a specific function is always accomplished within the required time period specified. DCD Tier 2, Revision 9, Section 7.1.3.2.7, states that the DCIS data communication functions are embedded within the Q-DCIS and the N-DCIS architectures. Safety internal and external communication protocols are deterministic. The RTIF-NMS and ATWS/SLC logic automatically initiates reactor trip and the SSLC/ESF, LD&IS, and VBIF logic automatically actuates the ESF that mitigate the consequence of DBEs. These automatic protection actions are implemented through 2/4 voting logic whenever one or more process variables reach their actuation setpoint. Variables are monitored and measured by each of the RTIF-NMS, ATWS/SLC, SSLC/ESF, and VBIF divisions. As documented in DCD Tier 2, Revision 9, Sections 7.2.1.3.5 and 7.3.5.3.5, the real-time performance of the Q-DCIS meets the requirements for the safety system trip and initiation response in conformance with BTP HICB-21. As part of the DAC closure process in DCD Tier 1, Revision 9, Section 3.2, the applicant will define the program cycle architecture to show how the deterministic character is achieved for each of the Q-DCIS platforms. Accordingly, the staff finds that the data communication design has proper deterministic character provisions.

7.1.5.3.3 Evaluation of Data Communication Systems Compliance with GDC

The staff reviewed the acceptance criteria for data communication systems in SRP Section 7.9 and SRP Appendix 7.1-A. Section 7.1.1.3 of this report evaluates the conformance of the Q-DCIS to the regulations and guidelines. The evaluation in this section will rely on applicable portions of the Section 7.1.1.3 evaluation and other evaluations specific to the data communication systems consistent with SRP Section 7.9.

GDC 1 requires quality standards and maintenance of appropriate records.
10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The staff evaluated whether GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed for the data communication systems, in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. The staff's evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1) and GDC 1 in Sections 7.1.1.3.3 and 7.1.1.3.6 of this report applies to the data communication systems. Therefore, the staff finds that the requirements of GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the data communication systems, in accordance with SRP

Appendix 7.1-A and Section 7-9. The review included the identification of those subsystems of the data communication systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Chapter 3 of this report. DCD Tier 1, Revision 9, Table 3.8-1, Items 1 and 3, include the ITAAC for the applicant to verify the EQ of safety electrical and digital I&C equipment. Accordingly, because the applicant has identified EQ programs consistent with the design bases for the data communication systems and the ITAAC for verification, the staff finds that the requirements of GDC 2 and 4 are adequately addressed.

The staff evaluated whether GDC 13 and 19 are adequately addressed for the data communication systems. GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. Section 7.1.1.3.6 of this report provides the evaluation of conformance with GDC 19 with the exception of data communication systems support functions necessary for operating the reactor. The applicant has identified interrelated processes to design the monitoring capability and control room controls. NEDE-33226P and NEDE-33245P, as part of a software life cycle process, define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing an HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room and remote shutdown panels to maintain the nuclear power unit in a safe condition during shutdown, including shutdown following an accident. Accordingly, based on the defined processes for designing the monitoring capability and the control room functions and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed for the data communication systems.

The staff evaluated whether GDC 13 and 19 are adequately addressed for the data communication systems that support protection system functions. As described above, Sections 3.2 and 3.3 of DCD Tier 1, Revision 9 include the DAC/ITAAC for verifying the software development and the HFE processes associated with the design and verification of controls and information displays for monitoring variables and systems in the control room. The DCD Tier 1 DAC/ITAAC are applicable to the data communication systems for protection systems. As described in Sections 7.2 and 7.3 of this report, the DAC/ITAAC include verifying the controls for manual initiation and control of functions in the control room necessary to support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Accordingly, based on the defined processes for designing the monitoring capability and the control room functions and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed for the data communication systems that support protection system functions.

The staff evaluated whether GDC 13 and 19 are adequately addressed for the data communication systems that support safe shutdown systems, information systems, and interlock logic important to safety, reactor control systems, and the DPS functions. As described above, Sections 3.2 and 3.3 of DCD Tier 1, Revision 9, include the DAC/ITAAC for verifying the software development and the HFE processes associated with the design of information displays for monitoring variables and systems and control room controls. The DCD Tier 1 DAC/ITAAC apply to the data communication systems for safe shutdown systems, information systems, and interlock logic important to safety, reactor control systems, and the DPS functions. As described in Sections 7.4, 7.5, 7.6, 7.7, and 7.8 of this report, the DAC/ITAAC include verification of the design and transmission of the variables and commands necessary to maintain the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems within prescribed operating ranges during plant shutdown. The DAC/ITAAC includes verification of instruments and controls within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including shutdown following an accident. Accordingly, based on the defined processes for designing the monitoring capability and the control room functions and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed for the data communication systems that support safe shutdown systems, information systems, and interlock logic important to safety, reactor control systems, and the DPS functions.

As described in the previous three paragraphs concerning general data communications and data communication systems for protection systems, safe shutdown systems, information systems, and interlock logic important to safety, reactor control systems, and the DPS functions, based on the defined processes for designing the monitoring capability and the control room functions and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed.

The staff evaluated whether GDC 21 is adequately addressed for the data communication systems. GDC 21 requires that protection systems be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. SRP Appendix 7.1-A states that GDC 21 is addressed for protection systems by conformance to the IEEE Std 603 criteria, except for Sections 5.4, 6.1, and 7.1. In addition, SRP Section 7.9 identifies that GDC 21 is addressed by conformance to RGs 1.22, 1.47, 1.53, and 1.118, and IEEE Std 379. DCD Tier 2, Revision 9, Section 7.1, describes the conformance of the DCIS to IEEE Std 603, which is evaluated in Section 7.1.1.3.10 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the design of the Q-DCIS complies with IEEE Std 603. In particular, DCD Tier 1, Table 2.2.15-2, Item 11, confirms Sections 5.7 and 6.5. DCD Tier 2, Revision 9, Table 7.1-1, identifies that the guidelines for periodic testing in RGs 1.22 and 1.118 apply to the DCIS. The bypassed and inoperable status indication conforms to the guidelines of RG 1.47. DCD Tier 2, Revision 9, Section 7.1.2.4, states that the DCIS conforms to the guidelines on the application for the single failure criterion in IEEE Std 379, as supplemented by RG 1.53. Section 7.1.1.3.3 of this report addresses the staff evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1). Based on the above, the the staff finds that the requirements of GDC 21 are adequately addressed for data communication systems.

Consistent with SRP Section 7.9, the staff also addressed compliance with GDC 21 through its review of the SDPs and design outputs and the applicant's conformance to RG 1.152. As discussed in Section 7.1.2 of this report, NEDE-33226P, in its description of the SDP, identifies appropriate output documentation with regard to data communications. DCD Tier 1, Revision 9, Section 3.2, provides the DAC/ITAAC to confirm the implementation of the SDP. Section 7.1.2 of this report provides further discussion of the SDPs. Based on the identified output documentation and its confirmation in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that GDC 21 and IEEE Std 7-4.3.2, as endorsed by RG 1.152, are adequately addressed with regard to SDPs.

The staff evaluated whether GDC 22 and the SRM to SECY-93-087, Item II.Q, are adequately addressed for the data communication systems. GDC 22 requires, in the pertinent part, that protection systems be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function. SRP Section 7.9 identifies that GDC 22 is addressed by the review of EMI/RFI susceptibility and that seismically exposed portions of the DCIS conform to GDC 2. SRP Section 7.9 also identifies that GDC 22 is addressed by conformance to IEEE Std 603, Section 5.6, and RG 1.75. As discussed in Section 7.1.1.3.10 of this report and in Items (11) and (13) in Section 7.1.5.3.1 of this report, as well as in the evaluation of GDC 2 above, the criteria related to EMI/RFI susceptibility and seismically exposed portions are adequately addressed for the DCIS. DCD Tier 2, Revision 9, Table 7.1-1, identifies that the guidelines in RG 1.75 apply to the Q-DCIS. DCD Tier 2, Revision 9, Section 7.1.6.4, describes the conformance of the Q-DCIS to IEEE Std 603, Section 5.6, which is evaluated in Section 7.1.1.3.10 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for verifying compliance with IEEE Std 603, Section 5.6.

Section 7.9 also identifies that the D3 assessment and conformance to the SRM to SECY-93-087, Item II.Q, should be reviewed for data communication systems that are part of protection and diverse systems in the review of GDC 22. The staff evaluated whether the DCIS functions were included in the staff's review of the D3 analysis for RPS and SSLC/ESF, as described in Section 7.1.3 of this report. The staff's evaluation of the D3 assessment and conformance to the SRM to SECY-93-087, Item II.Q, in Sections 7.1.3 and 7.1.1.3.7 of this report applies to the data communication systems. Based on the above, the the staff finds that the requirements of GDC 22 are adequately addressed.

The staff evaluated whether GDC 23 is adequately addressed for the data communication systems. GDC 23 requires that protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis. The staff's evaluation of conformance to GDC 23 in Section 7.1.1.3.6 of this report applies to the data communication systems. Therefore, the staff finds that the requirements of GDC 23 are adequately addressed.

The staff evaluated whether GDC 24 is adequately addressed for the data communication systems. GDC 24 requires that the protection system 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 of the protection system. GDC 24 also requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff's evaluation of conformance to GDC 24 in Section 7.1.1.3.6 of this report applies to the data communication systems. Therefore, the staff finds that the requirements of GDC 24 are adequately addressed.

The staff evaluated whether GDC 29 is adequately addressed for the data communication systems. GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. SRP Appendix 7.1-A states that GDC 29 is addressed by conformance, as applicable, to GDC 20-25 and GDC 28, which include verification of the design by DAC/ITACC. Sections 2.2.15, 3.2, 3.3, and 3.8 of DCD Tier 1, Revision 9, include the DAC/ITAAC for the applicant to verify that the data communication systems design implements these design criteria. The staff's evaluation of conformance to GDC 29 in Section 7.1.3.3.6 of this report applies to the data communication systems. Therefore, the staff finds that the requirements of GDC 29 are adequately addressed.

The staff evaluated whether 10 CFR 50.62 is adequately addressed for the data communication systems. As discussed in Section 7.8 of this report, the staff finds that the applicant addresses the 10 CFR 50.62 requirements, which require that the ARI system be diverse from the RPS, be designed to perform its function in a reliable manner, and be independent from the RPS and an SLC system to perform its function in a reliable manner. Accordingly, the staff finds that the requirements of 10 CFR 50.62 are adequately addressed.

The staff evaluated whether 10 CFR 50.34(f)(2)(v) is adequately addressed for the data communication systems. As described in Section 7.1.1.3.4 of this report, the staff evaluated compliance of the DCIS with 10 CFR 50.34(f)(2)(v) and finds it acceptable. This evaluation applies to the data communication systems. Accordingly the staff finds that the requirements of 10 CFR 50.34(f)(2)(v) are adequately addressed.

The staff evaluated whether 10 CFR 50.55a(h) is adequately addressed for the data communication systems. 10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995, which is evaluated in Section 7.1.1.3.10 of this report. The staff's evaluation of conformance to IEEE Std 603 in Section 7.1.3.3.10 of this report applies to the data communication systems. Therefore, the staff finds that the requirements of 10 CFR 50.55a(h) are adequately addressed.

The staff evaluated whether the applicant met the requirements of 10 CFR 52.47(b)(1). This regulation requires that the application for design certification contain 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 references the design certification has been constructed and will be operated in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. Section 7.1.5.3 and its subsections of this report address the ITAAC specific to the data communication systems. The staff evaluation of conformance to 10 CFR 52.47, "Contents of Applications; Technical Information," in Section 7.1.1.3.4 of this report applies to the data communication systems. Therefore, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed.

7.1.5.4 *Conclusion*

Based on the review of information documented in DCD Tier 2, Revision 9, Section 7.1.6.6.1; and DCD Tier 1, Revision 9, Tables 2.2.15-1, and 2.2.15-2, the staff concludes that the applicant adequately addresses the major design considerations for data communication systems. As discussed in Sections 7.1.1.3.1 through 7.1.1.3.10 of this report, as well as Section 7.1.5.3 above, the staff concludes that, for data communication systems, the applicant adequately addresses the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(v), 10 CFR 50.62, 10 CFR 52.47(b)(1), GDC 1, 2, 4, 13, 15, 19, 21, 22, 23, 24, 28, and 29; and the guidelines of the SRM to SECY-93-087. The staff also concludes that DCD Tier 2, Revision 9, identifies adequate high-level functional requirements, and DCD Tier 1, Revision 9, includes sufficient DAC/ITAAC to verify that the design is completed in compliance with the applicable requirements.

7.1.6 Secure Development and Operational Environment

7.1.6.1 *Regulatory Criteria*

RG 1.152, 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, 10 CFR Part 50, Appendix B, Criterion III, and IEEE Std 603-1991, Sections 5.6.3 and 5.9) for promoting high functional reliability, design quality, and a secure development and operational environment (SDOE) for use of digital computers in safety systems of nuclear power plants. SDOE in this context refers to protective actions taken against a predictable set of non-malicious acts (e.g., inadvertent operator actions, undesirable behavior of connected systems) that could challenge the integrity, reliability, or functionality of a digital safety system. RG 1.152, Revision 2 utilizes the waterfall life cycle phases to provide (1) a framework for establishing digital safety system, (2) SDOE guidance, and (3) criteria for acceptability, for the development of high quality safety systems. By committing to RG 1.152, Revision 2 in the DCD, the development of the Critical Digital Assets (CDAs) is evaluated to the criteria for securing the development process and providing reliable secure operational environment features within the design for the identified life cycle phases, which consists of the following phases:

- Concepts
- Requirements
- Design
- Implementation
- Test
- Installation, Checkout, and Acceptance Testing
- Operation
- Maintenance
- Retirement

The staff's acceptance of system SDOE design features is based on (1) confirming that the appropriate CDA analysis and grouping are made, (2) confirming that appropriate secure operational environment design elements are integrated into project specific plans, (3) verifying that the secure operational environment elements of the plans are followed, (4) confirming that the process produces acceptable secure operational environment design outputs, and (5) validating that an effective and responsive secure operational environment is utilized throughout all phases to protect process integrity.

Cyber security to address malicious events is under the purview of 10 CFR 73.54. Section 1.7 of NEDE-33295P states that while GEH's Cyber Security Program Plan and Cyber Security Program may be used to demonstrate compliance to aspects of 10 CFR 73.54, conformance to 10 CFR 73.54 is the responsibility of the COL applicant to demonstrate compliance, not GEH. Thus the staff did not evaluate the portion of NEDE-33295P that addresses aspects of 10 CFR 73.54 for compliance to this requirement. Compliance with 10 CFR 73.54 will be addressed by the COL applicant. Therefore, the ESBWR design certification does not include compliance with 10 CFR 73.54.

In the context of this safety evaluation report, instances in which the DCD or its referenced documents uses the word "cyber security," the evaluation will be based on those SDOE features that address non-malicious acts.

7.1.6.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 7.1.6.6.1.28, states that the cyber security measures included in RG 1.152 are evaluated and incorporated into the Q-DCIS design in accordance with NEDE-33295P. Coverage is extended to all systems and components determined to be CDAs. Cyber security design functionality is directly integrated into the project specific plans developed under NEDE-33226P and NEDE-33245P.

NEDE-33295P is a high level document defining the requirements for the development and management of an effective SDOE program for the applicant and its ESBWR product. This document is a top-tier design basis and high-level implementation guide for the ESBWR SDOE Program, in accordance with RG 1.152.

NEDE-33295P provides information on the system design cyber security as well as other cyber security material that is outside the scope of the design certification, as explained above. This review only bases its regulatory findings on Regulatory Positions C.2.1 through C.2.6 of RG 1.152, Revision 2 (i.e., Concepts through Installation, Checkout, and Acceptance Testing of the development of the CDA SDOE design). Compliance with Regulatory Positions C.2.7 through C.2.9 of RG 1.152, (i.e., Operation; Maintenance; and Retirement phases) will be addressed by the COL applicant. In addition, NEDE-33226P and NEDE-33245P address other guidance for system development, including the remainder of RG 1.152.

The applicant has not included any specific SDOE project plans or design outputs within the scope of the design certification. Instead, NEDE-33295P provides template information for properly integrating SDOE elements into the design of all CDAs. This document also specifies the processes required to be in place during all development activities to maintain a secure development environment during the development and other pre-operational life cycle activities. These requirements are binding upon all components of the applicant's source of supply, including vendors and sub contractors.

The DAC/ITAAC in DCD Tier 1, Revision 9, Section 3.2, are aligned to project activities, to confirm the proper integration of functions and features that support a SDOE into the system life cycle. In addition, the DAC/ITAAC confirms the completion of these activities and conformance of the products to the processes described in NEDE-33295P and the guidelines of RG 1.152.

7.1.6.3 *Staff Evaluation*

7.1.6.3.1 Review Method for SDOE Design

The staff examines the software life cycle planning, implementation, and design outputs for elements of features and functions that support a SDOE. This information can be organized as described in RG 1.152, Section C.2.

The SDOE integration model described in RG 1.152, Section C.2, Regulatory Positions 2.1 through 2.9, is very similar to the waterfall life cycle described in BTP HICB-14. The secure development environment program should be integrated into the overall physical access control, intellectual property protection, and quality assurance programs in place at the applicant's facility. The staff will examine the applicant submissions to ensure the integration into current and planned practices already identified for system development.

RG 1.152 applies to safety systems. An applicant can make a determination to apply this RG to all CDAs. If this determination is made, then RG 1.152 guidance should be applied to all CDAs in a graded approach. The SDOE requirements for a specific life cycle phase should be commensurate with the risk and magnitude of the harm resulting from unauthorized and inappropriate access, use, disclosure, disruption, or destruction of the applicable digital system.

7.1.6.3.2 General Evaluation of SDOE Activities

In NEDE-33295P, the applicant states that it conforms to RG 1.152 for SDOE activities. This document describes the plan for developing the program that implements the SDOE guidance provided by RG 1.152 and BTP HICB-14. This LTR works in conjunction with the software development LTRs (NEDE-33226P and NEDE-33245P) to implement an overall SDOE system design. The two software LTRs provide the overall development guidance in accordance with BTP HICB-14. NEDE-33295P provides guidance to enhance and modify the overall system development process with functionality that supports SDOE. The staff's review of NEDE-33295P evaluated whether this LTR provides this information and gives adequate direction for implementing the SDOE processes in the project-specific plans.

The applicant identifies specific life cycle phases for this process. These life cycle phases are similar to those identified in RG 1.152, as well as those in BTP HICB-14. The reviewed activities in each phase are consistent with the guidance provided in RG 1.152. The applicant refers to the concepts phase as the planning phase. In addition, the operations phase and the maintenance phase are combined into one operations and maintenance phase. The operations and maintenance phase, as well as the retirement phase, are outside the scope of this review. Section 7.1.6.3.5 of this report documents the staff's review of the life cycle phases.

The applicant states that it will use the process outlined in NUREG/CR–6847, "Cyber Security Self-Assessment Method for U.S. Nuclear Power Plants (not publicly available)"; this process will be used to determine which systems or groupings of systems will be considered CDAs. The applicant has expanded the use of the basic process by applying it to a conceptual design rather than just an as-built system. This process will be used iteratively and increase the quality and completeness of the system design. The staff finds the expanded use of NUREG/CR–6847 acceptable.

7.1.6.3.3 Evaluation of Compliance with Regulations

The staff has reviewed NEDE-33295P for compliance with the requirements of GDC 21 and IEEE Std 603-1991, Sections 5.6.3, and 5.9 as incorporated by reference in 10 CFR 50.55a(h) for promoting high functional reliability, design quality, and a secure development and operational environment for use of digital computers in safety systems of nuclear power plants. Designing in features to protect the protection system against a predictable set of non-malicious acts (e.g., inadvertent operator actions, undesirable behavior of connected systems) provides enhanced assurance of high functional reliability. Taking the same actions with systems which could adversely impact the ability of a safety system to perform its safety function also provides enhanced assurance of high functional reliability. The guidance in RG 1.152 provides general guidance by which the applicant can support high functional reliability. The use of a NUREG/CR–6847 method provides a means to identify all of those systems that may be affected by non-malicious events, including nonsafety systems. Based on the applicant's commitment to follow the applicable guidance in RG 1.152, as detailed in NEDE-33295P and confirmed in this review, the staff finds that the applicant's submission meets the requirements of GDC 21 and Sections 5.6.3 and 5.9 of IEEE Std 603-1991 and therefore 10 CFR 50.55a(h).

The staff also reviewed NEDE-33295P for compliance to the quality assurance requirements of 10 CFR Part 50, Appendix B. Quality assurance comprises all those planned and systematic actions necessary to provide adequate assurance that a system will perform satisfactorily in service. In particular, quality assurance includes quality control. An adequate secure development environment program is important to control the quality of the systems being developed under this program and to protect them from the effects of cyber events. Based on the applicant's commitment to follow the applicable guidance in RG 1.152 for quality assurance, as detailed in NEDE-33295P and as documented in DCD Tier 2 in NEDE-33245P, the NRC staff finds that for cyber security, the applicant voluntarily meets the requirements for a sufficient quality assurance program in accordance with 10 CFR Part 50 Appendix B.

7.1.6.3.4 Evaluation of Deviations from Guidelines and Standards

In Appendix A, "Conformance Review," of NEDE-33295P, the applicant documented the conformance with RG 1.152 and also addressed some deviations. With the exceptions of the stated deviations, NEDE-33295P specifies conformance to the criteria and guidance contained in RG 1.152. The applicant's stated deviations from applicable RGs and industry standards are summarized and evaluated below.

(1) BTP HICB-14

The applicant deviates in the use of metrics to monitor the development process.

Section 7.1.2 of this report addresses this issue in its review of the software development methodology, which the staff finds acceptable.

(2) RG 1.152

The applicant excludes IEEE Std 12207.0-1996 and IEEE Std 603-1998, from stated conformance. The applicant describes this nonconformance as follows:

> IEEE Std 12207.0-1996 is not directly referenced. However, IEEE Std 1074-1995 is directly referenced by RG 1.173, and is therefore used instead of IEEE Std 12207. IEEE Std 1074 covers similar topics and is the committed reference. IEEE Std 603-1998 addresses criteria for safety systems but is not within the scope of the SMPM and SQAPM because it does not provide guidance on software design and software quality assurance.

Since IEEE Std 12207 has not been endorsed by an RG, and the applicant followed IEEE Std 1074, which is endorsed by RG 1.173, this deviation is acceptable.

RG 1.152 applies only to safety-related systems. The applicant deviates by expanding the scope to all CDAs. This deviation is acceptable.

Parts of RG 1.152, Section C.2 are scoped to the licensee. This document does not address these actions. This deviation is acceptable.

The NRC staff reviewed the deviations from RG 1.152 and finds them acceptable.

7.1.6.3.5 Evaluation of Life Cycle Phase Activities

The primary purpose of NEDE-33295P is to provide guidance and direction to enhance and modify the project-specific plans to include adequate SDOE functionality. Therefore, the staff reviewed the expected content of each life cycle phase activity listed in this plan against RG 1.152. The staff also considered applicant deviations in the review of these activities.

The NRC staff evaluated NEDE-33295P life cycle phase activities as follows:

(1) Concepts Phase

The applicant refers to this phase at the Planning phase. In this phase, the CDAs are identified in a NUREG/CR–6847 process. Communication pathways and interfaces are defined. System design vulnerabilities are listed. Technologies to mitigate the vulnerabilities are identified. These activities are basically the same as those identified in RG 1.152. Some activities cross the boundary between Concepts and Requirements. However, this is acceptable because the information is appropriately developed and available when needed.

The activities in this phase are acceptable.

(2) Requirements Phase

System architecture requirements unique to each identified CDA are developed and defined. The requirements are directly driven by the outputs of the Planning phase. These requirements are integrated into the overall system development requirements.

Network specific architecture issues are identified at this stage. Concepts for system configuration, access control and similar protection level driven items are addressed. These items will be integrated into the overall Hardware and Software Specification. System interfaces will be addressed in further detail. COTS software and previously developed software requirements will be derived and evaluated. Identified V&V tasks will be evaluated. Any additional requirements in this area will be integrated into the overall project-specific V&V plans. The activities identified by the applicant meet the threshold identified in RG 1.152. The activities in this phase are acceptable.

(3) Design Phase

The design phase addresses the concepts of confidentiality, integrity and availability. The activities are directly integrated into the SMPM identified SDOE activities. The vulnerability assessment is taken from the NUREG/CR–6847 review and integrated into the SMPM and SQAPM activities and derived project-specific plans. Physical and logical access as well as interfaces between digital assets and other networks is addressed in accordance with RG 1.152. Additional SDOE requirements to address access to CDAs are addressed. This phase is in line with the activities detailed in RG 1.152. The activities in this phase are acceptable.

(4) Implementation Phase

The secure coding practices required by the SMPM are followed in this phase. Coordination between the SMPM and SQAPM commitments are specifically discussed in this phase. Unique issues of COTS software and its vulnerability evaluation are discussed. The general secure coding practices and procedures already present and identified by RG 1.152 are reemphasized in this phase. Particular attention is given to the difficult problems of securing COTS software. The NRC staff has reviewed this high level process description and finds that the activities in this phase are acceptable.

(5) Test Phase

The test activities are generally covered in the SMPM and SQAPM. Specific issues related to SDOE are driven by RG 1.152 and the secure operational environment hazard analysis and related activities. Scanning and other actions will be performed to verify the functions and features that a support secure operational environment as designed into the system are adequate. An option to perform more in-depth scanning based on NUREG/CR–6847 procedures is also discussed. The majority of the testing activities are covered and approved in the SMPM and SQAPM documents. The implementation of these documents through their derivative project-specific plans provide assurance that sufficient testing will take place to validate the design functions and features that support a secure operational environment for each CDA. Based on NRC staff's review and the conformance with the guidance on RG 1.152, the activities in this phase are acceptable.

(6) Installation, Checkout and Acceptance Testing Phase

The applicant refers to this phase as simply the Installation phase. The SMPM governs the general installation phase activities. Additional tests and procedures to validate the functions and feature that support a secure operational environment in the CDA will be integrated into the appropriate project-specific plans. The activities specifically listed in this section are equivalent or surpass the information provided in RG 1.152. The activities in this phase are acceptable.

The NRC staff review of the NEDE-33295P finds that this document provides adequate, high level guidance to support the design and implementation of functions and features that support a secure operational environment. This determination is based on the identification of the activities specified in RG 1.152 to establish and maintain a SDOE.

7.1.6.3.6 Evaluation of SDOE Activities DAC/ITAAC

The applicant has chosen to use the DCD Tier1 DAC/ITAAC process to allow completion of both system and project-specific SDOE design activities, as well as the completion of verifiable activities through the project-specific life cycle phases up to fuel load.

The necessary DAC/ITAAC items are derived from the RG 1.152 process and resulting documents. The DAC/ITAAC documented in DCD Tier 1, Revision 9, Section 3.2 confirm and verify that (1) the appropriate CDA analysis and grouping are made, (2) appropriate secure operational environment design elements are integrated into project specific plans, (3) the secure operational environment design elements of the plans are followed, (4) the process produces acceptable secure operational environment design outputs, and (5) an effective and responsive secure development environment plan is being utilized throughout all phases to protect process integrity.

7.1.6.4 *Conclusion*

The applicant has identified deviations from RG 1.152 guidance and applicable regulations. The staff reviewed these deviations and finds them to be acceptable. The applicant's proposed design SDOE activities address the relevant requirements of 10 CFR 50.55a(h); GDC 21; and

10 CFR Part 50 Appendix B. The NRC staff finds that the applicant's SDOE design activities are consistent with RG 1.152 and associated regulatory guidance protection against a predictable set of non-malicious acts that could challenge the integrity, reliability, or functionality of a digital safety system.

The applicant has provided sufficient information on SDOE provisions in DCD Tier 1, Revision 9, Section 3.2; DCD Tier 2, Revision 9; and NEDE-33295P to provide assurance that the defined process will sufficiently integrate functions and features that support a secure operational environment into the project specific software development plans. This integration will result in high quality safety system software with appropriate functions and features to address a predictable set of non-malicious acts.

Based on the review of DCD Tier 1, Revision 9, Section 3.2; DCD Tier 2, Revision 9; and NEDE-33295P documentation, the staff concludes that the SDOE activities are acceptable.

7.2 Reactor Trip System

7.2.1 Regulatory Criteria

The objective of the review of DCD Tier 1, Section 2.2, and DCD Tier 2, Section 7.2, is to confirm that the RTS satisfies regulatory acceptance criteria, guidelines, and performance requirements. The review of the I&C aspects of the RTS includes the RPS, the NMS, and the SPTM functions. The RTS detects a plant condition that initiates rapid insertion of control rods to shut down the reactor in situations that could result in unsafe reactor operations. This action prevents or limits fuel damage and system pressure excursions, minimizing the release of radioactive material.

Acceptance criteria in SRP Section 7.2, Revision 5, for the RTS, hence the RPS as discussed below, are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(v); 10 CFR 50.34(f)(2)(xxiii); 10 CFR 52.47(b)(1); and GDC 1, 2, 4, 10, 13, 15, 19, 20-25, and 29. The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152, and the SRM to SECY-93-087.

7.2.2 Summary of Technical Information

7.2.2.1 *Reactor Protection System Description and Architecture*

7.2.2.1.1 Reactor Protection System Description

The RPS is designed to provide the capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures. The RPS logic will not result in a reactor trip when one entire division of channel sensors is bypassed or when one of the four automatic RPS trip logic systems is out of service (with any three of the four divisions of safety power available). This is accomplished through the combination of fail-safe equipment design, the redundant sensor channel trip decision logic, and the redundant 2/4 trip systems output scram logic. The RPS is classified as a safety system. The RPS electrical equipment is classified as seismic Category I and will be environmentally and seismically qualified.

The RPS logic design will be such that it initiates reactor trip signals within individual sensor channels when any one or more of the conditions listed below exists during reactor operation. A reactor scram results if system logic is satisfied. The following is a list of the process conditions and, in parentheses, the systems monitoring the process conditions:

- High drywell pressure (CMS)
- Turbine stop valve closure (RPS)
- Turbine control valve fast closure (RPS)
- NMS - monitored SRNM and APRM conditions exceed acceptable limits (NMS)
- Reactor vessel pressure high (NBS)
- RPV water level low (Level 3) decreasing (NBS)
- RPV water level high (Level 8) increasing (NBS)
- MSIVs closure (run mode only) (NBS)
- Low-low CRD hydraulic control unit (HCU) accumulator charging header pressure (CRD)
- Suppression pool temperature high (CMS)
- High condenser pressure (RPS)
- Power generator bus loss (loss of feedwater flow) (run mode only) (RPS)
- High simulated thermal power (feedwater temperature biased) (NBS and NMS)
- Feedwater temperature exceeding allowable simulated thermal power versus feedwater temperature domain (NBS)
- Operator-initiated manual scram (RPS)
- Reactor mode switch in “shutdown” position (RPS)

7.2.2.1.2 Reactor Protection System Architecture

Four instrument channels are provided for each process variable being monitored, one for each RPS division. When more than four sensors are required to monitor a variable, the output of the sensors are combined into only four instrument channels. The logic in each division is asynchronous with respect to the other divisions. The RPS is implemented with two communication methodologies: “point-to-point” optical fiber inter-divisional communication and a shared memory data communication ring network. Point-to-point communication is limited to trip and bypass information. Point-to-point fiber is also used for functional trip logic units (TLUs) to output logic units (OLUs), RPS to NMS and RPS to SSLC/ESF communication. The shared memory data communication ring network can read the entire shared memory on the CIMs card and write only to a designated portion of the CIMs card. The data on a data communication ring are actively transported between one chassis transmitter and another’s receiver until all nodes are updated. Two “counter rotating” data communication rings are within each division; therefore, no single failure will prevent data transmission.

Equipment within a sensor channel consists of sensors (transducers or switches), the DTM, and multiplexers. The sensors within each channel detect abnormal operating conditions and send analog (or discrete) output either directly to the RPS cabinets or to the RMUs within the associated division of the Q-DCIS. The RMUs within each division perform analog-to-digital conversion and signal processing and then send the digital or digitized analog output values of the monitored variables to the DTM for trip determinations within the associated RPS sensor channel in the same division. The DTM in each sensor channel compares individual monitored variable values with trip setpoint values and, for each variable, sends a separate trip/no trip output signal to the TLUs in the four divisions of trip logic.

Equipment within an RPS division of trip logic consists of TLUs, manual switches, bypass units (BPUs), and OLUs. The TLUs perform the automatic scram initiation logic, checking for 2/4 coincidence of trip conditions in any set of instrument channel signals coming from the four divisions of DTMs or when an NMS isolated digital trip signal (voted 2/4 in the NMS TLU) is received. The automatic scram initiation logic for any trip is based on the reactor operating mode switch status, channel trip conditions, NMS trip input, and bypass conditions. Each TLU, besides receiving the signals described above, also receives digital input signals from the BPUs and other control interfaces in the same division. The BPUs perform bypass and interlock logic for the division of channel sensors bypass and the division TLU bypass. Each BPU sends a separate bypass signal for the four channels to the TLU in the same division for channel sensors bypass. Each RPS BPU also sends the TLU bypass signal to the OLU in the same division.

The OLUs perform division trip, seal-in, reset, and trip test functions. Each OLU receives bypass inputs from the RPS BPUs, trip inputs from the TLU of the same division, and manual inputs from switches within the same division. Each OLU provides trip outputs to the trip actuators.

Equipment within a division of trip actuators includes load drivers for automatic primary scram and initiation of backup scram. The RPS includes two physically separate and electrically independent divisions of trip actuators receiving inputs from the four divisions of OLU. The operation of the load drivers is such that a trip signal on the input side creates a high impedance, current-interrupting condition on the output side. The output side of each load driver is electrically isolated from its input signal. The load driver outputs are arranged in the primary scram logic circuitry, which is between the scram solenoids and scram solenoid 120-volts ac power source. When in a tripped state, the load drivers cause the scram solenoids (scram initiation) to de-energize. The load drivers within a division interconnect with the OLU of all other divisions to form a special arrangement (connected in series and in parallel in two separate groups) that result in 2/4 scram logic. Reactor scram occurs if load drivers associated with any two or more divisions receive trip signals from the OLUs.

Load drivers are also used for backup scram actuators, scram-follow initiation, and scram reset permissive actuators. When in a tripped state, the load drivers for backup scram cause the air header dump valve solenoids (air header dump initiation) to energize. The load drivers of the backup scram are arranged in a 2/4 configuration similar to that described above for the primary scram load drivers. Backup scram is diverse in power source and function from primary scram.

Equipment within a division of manual scram controls includes manual switches, contacts, and relays that provide an alternate, diverse, manual means to initiate a scram and air header dump. Each division's manual scram function controls the power sources to the same division of scram logic circuitry for scram initiation and division of scram logic circuitry for air header dump initiation. One of the two divisions of scram logic circuitry distributes Division 1 safety 120-volts ac power to the A solenoids of the HCUs. The other division of scram logic circuitry distributes Division 2 safety 120-volts ac power to the B solenoids of the HCUs. The HCUs (which include the scram pilot valves and the scram valves) and the air header dump (backup scram) valves are themselves, components of the CRD system.

7.2.2.2 *Neutron Monitoring System Description*

DCD Tier 2, Revision 9, Section 7.2.2, describes the NMS. The NMS monitors reactor core thermal neutron flux from the startup source range to beyond rated power and provides trip signals initiating reactor scrams under excessive neutron flux or excessive rates of change in neutron flux (short period) conditions. The NMS comprises the following subsystems:

- SRNM
- PRNM
- AFIP
- MRBM

The SRNM and PRNM subsystems, discussed below, are safety systems. The PRNM subsystem includes the LPRM, APRM, and OPRM functions. The AFIP subsystem and the MRBM, discussed in Section 7.7 of this report, are nonsafety systems.

7.2.2.2.1 Startup Range Neutron Monitor Subsystem

DCD Tier 2, Revision 9, Sections 7.2.1.2.4 and 7.2.2, describe the SRNM. The SRNM is designed as a safety subsystem generating trip signals to prevent fuel damage in the event of any abnormal reactivity insertion transients (while operating in the startup power range). The trip signal results either from an excessively high neutron flux level or an excessive rate of neutron flux increase (a short reactor period). The setpoints of these trips are such that, under the worst reactivity insertion transients, fuel integrity is always protected. DCD Tier 2, Revision 9, Table 7.2-2, provides the SRNM trip and rod block functions. DCD Tier 2, Revision 9, Table 7.2-3, provides the SRNM trip signals. The trip setpoints are adjustable and are determined using the setpoint methodology in NEDE-33304P.

7.2.2.2.2 Power Range Neutron Monitor

7.2.2.2.2.1 *Local Power Range Monitor*

The LPRM is designed to monitor the local power level and to provide a sufficient number of LPRM signals to the APRM system to fulfill the safety design basis for the APRM. The LPRM is qualified to operate under DBAs and abnormal environmental conditions.

The LPRM has the following design characteristics:

- Provides signals to the APRM that are proportional to the local neutron flux at various locations within the reactor core

- Provides signals to alarm high or low local thermal neutron flux

- Provides signals proportional to the local neutron flux to drive indicators and displays and for the PCF used for operator evaluation of power distribution

- Provides signals proportional to the local neutron flux for use by other interface systems such as the RC&IS, for the rod block monitoring function

7.2.2.2.2.2 *Average Power Range Monitor*

DCD Tier 2, Revision 9, Section 7.2.2.1.3.1, specifies the following safety design bases for the APRM:

- The functional requirements specify that, under the worst permitted input LPRM bypass conditions, the APRM is capable of generating a timely trip signal in response to excessive average neutron flux increases to prevent fuel damage.

- The system is designed to produce a safety simulated thermal power signal to the RPS to allow that system to support reactor power scram bypass requirements.

- The APRM provides information for monitoring the average power level of the reactor core in the power range. The APRM is capable of generating a trip signal to scram the reactor in response to excessive and unacceptable neutron flux increase to prevent fuel damage. Such a trip signal includes a trip from the simulated thermal power signal, representing the APRM flux signal through a time constant representing the actual fuel time constant. The resulting simulated thermal power signal accurately represents core thermal (as opposed to neutron flux) power and the heat flux through the fuel.

- Scram functions are assured when the minimum LPRM input requirement to the APRM is fulfilled. If this requirement cannot be met, an inoperative channel trip signal is generated. Independence and redundancy requirements are incorporated into the design and are consistent with the safety design basis of the RPS.

Additional design characteristics of the APRM include the following:

- Provides continuous indication of average reactor power (neutron flux) from 1 percent to 125 percent of rated reactor power, which overlaps with the SRNM range.

- Provides interlock logic signals for blocking further rod withdrawal to avoid an unnecessary scram actuation.

- Provides a simulated thermal power signal derived from each APRM channel, which approximates the heat dynamic effects of the fuel.

- Provides a continuously available LPRM/APRM display for detection of any neutron flux oscillation in the reactor core.

7.2.2.2.2.3 *Oscillation Power Range Monitor*

DCD Tier 2, Revision 9, Section 7.2.2.1.4.1, specifies the following safety design bases for the OPRM:

- Under the worst permitted input LPRM bypass conditions, the OPRM is capable of generating a timely trip signal in response to core neutron flux oscillation conditions and thermal-hydraulic instability to prevent violation of the thermal safety limit.
- The OPRM provides monitoring and protection function for core-regional and core-wide neutron flux oscillation monitoring using the LPRM signals sent to the associated APRM channel in which the OPRM channel resides. The OPRM is capable of generating a timely trip signal to scram the reactor in response to excessive and unacceptable neutron flux oscillation to prevent fuel damage. Scram functions are ensured when the minimum LPRM input requirement to the OPRM is fulfilled.
- The OPRM provides nonsafety core flux oscillation information for the plant computer functions and MCR display and alarms when the OPRM is inoperative or has an insufficient number of LPRM inputs.

7.2.2.3 *Suppression Pool Temperature Monitor*

The SPTM provides suppression pool temperature data for automatic scram and automatic suppression pool cooling initiation when established high temperature limits are exceeded. In addition, the SPTM subsystem provides suppression pool temperature data for operator information and recording and for post accident conditions of the suppression pool. The SPTM hardware is redundantly powered by the appropriate dual divisional uninterruptible 120-volts ac power sources, either of which can support the SPTM function.

The sensor electrical wiring, encapsulated in pliable, grounded sheathing, is terminated in wetwell-sealed, moisture-proof junction boxes for easy sensor replacement or maintenance during a plant outage. The temperature sensor wiring from the wetwell junction boxes is directed through the suppression pool divisional instrument penetrations to the four-divisional Q-DCIS RMUs and the DPS RMUs.

7.2.3 Staff Evaluation

The staff reviewed the RPS in accordance with SRP Section 7.2. The staff also used acceptance criteria in SRP Section 7.1, SRP Table 7-1, SRP Appendix 7.1-A, and SRP Appendix 7.1-C, as directed by SRP Section 7.2. Section 7.1.1.1 of this report describes the acceptance criteria listing used as the basis for the staff's review of the RPS. SRP Section 7.2 highlights specific topics that should be emphasized in the RPS review and are addressed in Section 7.2.3.1 of this report. The staff included the review of the DCD Tier 1, Revision 9, DAC/ITAAC during the review of this section because of their significant role in determining the RPS conformance to all requirements.

As described in Section 7.1.1.3.1 of this report, the DCD does not provide the required RPS system design information to comply with IEEE Std 603. Instead, the applicant has included the DAC/ITAAC in DCD Tier 1, Revision 9, Section 2.2.15, to confirm that the completed RPS design complies with IEEE Std 603. DCD Tier 1, Section 2.2.15, also includes a DAC/ITAAC applicability table (Table 2.2.15-1), which identifies the applicability of the IEEE Std 603 criteria DAC/ITAAC to the RPS. The staff has accepted the DAC approach to addressing compliance with IEEE Std 603. The staff's evaluation of conformance to IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the RPS.

In RAI 7.1-99, the staff asked the applicant to clarify the applicability of IEEE Std 603 criteria in a consistent manner throughout DCD Tier 2, Chapter 7. The applicant submitted a response to

RAI 7.1-99, along with responses to RAIs 7.1-100, 7.1-101, and supplemental RAI 4.3-265 S01, all of which are incorporated in DCD Revision 8. Section 7.1.1.3.1 of this report provides additional discussion on the resolution of these RAIs. With regard to the staff's evaluation of the RPS, the applicant significantly revised DCD Tier 2, Revision 9, Tables 7.1-1 and 7.1-2, to clearly identify the applicability of IEEE Std 603 criteria for each RPS subsystem. Concurrently, the applicant revised or removed many references to IEEE Std 603 criteria from the discussions of RPS subsystems in DCD Tier 2, Revision 9, Section 7.2 to address the consistency concerns. Accordingly, statements regarding the applicability of IEEE Std 603 criteria provided in the staff's SER with open items are updated or removed from the remainder of Section 7.2.3 of this report to be consistent with DCD Revision 9.

7.2.3.1 *Evaluation of Reactor Protection System Conformance with Acceptance Criteria - Major Design Considerations*

SRP Section 7.2 lists the following eight major design considerations that the review should emphasize:

(1) Design Basis (IEEE Std 603, Section 4)

The staff evaluated the RPS design basis to determine whether IEEE Std 603, Section 4, is adequately addressed using SRP Appendix 7.1-C, Section 4. For completeness, the SRP states, "As a minimum each of the safety system design basis aspects identified in IEEE Std 603, Sections 4.1 through 4.12 should be addressed." Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 4 is adequately addressed based on its inclusion in the safety system design basis and the verification of applicable criteria in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the RPS. Also, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for the relevant criteria of Section 4 are applicable to the RPS and all subsystems.

DCD Tier 2, Revision 9, Section 7.2.1.2.4.2, identifies individual parameters that determine when, in a particular condition or extreme, the RPS automatically initiates a reactor scram. As mentioned previously, NEDE-33226P and NEDE-33245P, as part of the software life cycle process, define a process by which plant performance requirements, including response times, under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistently with this process and produce acceptable design outputs. Accordingly, the staff finds that Section 4 is adequately addressed for the RPS.

(2) Single Failure Criterion (IEEE Std 603, Section 5.1)

The staff evaluated whether the single failure criterion in IEEE Std 603, Section 5.1, is adequately addressed for the RPS. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 5.1 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation is applicable to the RPS. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 5.1 apply to the RPS and all subsystems. Accordingly, the staff finds that Section 5.1 is adequately addressed for the RPS.

(3) Quality of Components and Modules (IEEE Std 603, Section 5.3)

The staff evaluated whether the quality criterion, IEEE Std 603, Section 5.3, is adequately addressed for the RPS. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 5.3 is adequately addressed based on the inclusion of IEEE Std 7-4.3.2, Section 5.3, in the safety system design basis and the verification of the software development activities in the DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC. In addition, the applicant stated that the quality assurance program conforms to GDC 1. Chapter 17 of this report addresses the evaluation of the adequacy of the quality assurance program. These evaluations apply to the RPS. DCD Tier 2, Revision 9, Section 7.1.6.6.1.4, also discusses the applicability of this criterion to the Q-DCIS design. Accordingly, based on the applicant's use of an acceptable software development process, as evaluated in Section 7.1.2.3 of this report, and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.3 of IEEE Std 7-4.3.2 and Section 5.3 of IEEE Std 603 are adequately addressed for the RPS.

(4) Independence (IEEE Std 603, Sections 5.6 and 6.3)

The staff evaluated whether the independence-related criteria, IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed for the RPS. Section 7.1.1.3.10 of this report evaluates IEEE Std 603, Section 5.6, and the staff finds that Section 5.6 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the RPS. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 5.6 apply to the RPS and all subsystems. Accordingly, the staff finds that Section 5.6 is adequately addressed for the RPS.

The staff evaluated conformance with IEEE Std 603, Section 6.3. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 6.3 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the RPS. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 6.3 apply to the RPS and all subsystems. Accordingly, the staff finds that Section 6.3 is adequately addressed for the RPS.

(5) Diversity and Defense-in-Depth

The staff evaluated whether the RPS has adequate D3. The RPS should incorporate multiple means for responding to each event discussed in DCD Tier 2, Revision 9, Chapter 15. At least one pair of these means for responses to each event should have the property of signal diversity (i.e., the use of different sensed parameters to initiate protective action, in which any of the parameters may independently indicate an abnormal condition, even if the other parameters are sensed incorrectly (see NUREG/CR–6303). NEDO-33251 states conformity to NUREG/CR–6303. DCD Tier 1, Revision 9, Table 2.2.14-4, requires the applicant to perform FMEAs of the safety protection system platforms to validate the DPS functions. The staff evaluation of conformance to D3 in Section 7.1.3.3 of this report applies to the RPS. Accordingly, the staff finds that D3 is adequately addressed for the RPS.

(6) System Testing and Inoperable Surveillance (IEEE Std 603, Sections 5.7, 5.8, and 6.5)

The staff evaluated whether the criteria related to system testing and inoperable surveillance, IEEE Std 603, Sections 5.7, 5.8, and 6.5, are adequately addressed. Section 7.1.1.3.10 of this report evaluates IEEE Std 603, Section 5.7, and the staff finds that Section 5.7 is adequately addressed based on the inclusion of IEEE Std 603, Section 5.7, in the design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the RPS. IEEE Std 603, Section 5.7 states that the capability for testing and calibration of safety system equipment shall be provided during power operation and shall duplicate, as closely as practicable, performance of the safety function. The applicant identifies two exceptions to this criterion in the RPS as follows:

> 1) confirm operation of MSIV and turbine stop valve limit switches and 2) independent functional testing of the air header dump valves during each refueling outage (not operation) and operation of at least one valve can be confirmed following each scram.

The RPS can be tested in overlapping segments when testing one safety function. The extent of test and calibration capability provided depends on whether the design meets the single failure criterion. SRP Appendix 7.1-C states that any failure that is not detectable must be considered concurrently with any random postulated, detectable, single failure. DCD Tier 1, Revision 9, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.1, for the applicant to perform an analysis, or FMEA, that confirms that the requirements of the single failure criterion are satisfied for the RPS. DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 5.7 apply to the RPS and all subsystems. Accordingly, the staff finds that Section 5.7 is adequately addressed for the RPS.

The staff evaluated whether IEEE Std 603, Section 5.8, is adequately addressed. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that the criterion is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 3.3, DAC/ITAAC. This evaluation applies to the RPS. In addition, Section 5.8, "Information Displays," is part of system testing and inoperable surveillance. DCD Tier 2, Revision 9, Chapter 18, describes the HFE design process to design information displays, which is evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification applies to the RPS and includes verifying the inventory of displays for manually controlled actions, system status indications, and indications of bypasses. Accordingly, the staff finds that Section 5.8 is adequately addressed for the RPS.

The staff evaluated conformance with IEEE Std 603, Section 6.5. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 6.5 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the RPS. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 6.5 apply to the RPS and all subsystems. Accordingly, the staff finds that Section 6.5 is adequately addressed for the RPS.

(7) Use of Digital Systems (IEEE Std 7-4.3.2)

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed for the RPS. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2 concerning the use of digital systems. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in SRP Appendix 7.1-D. The staff evaluation of conformance to IEEE Std 7-4.3.2 in Section 7.1.1.3.10 of this report applies to the RPS.

NEDE-33226P and NEDE-33245P describe the software development activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff evaluation of software development activities. Accordingly, the staff finds that use of digital systems is adequately addressed for the RPS.

(8) Setpoint Determination

Section 7.1.4 of this report evaluates the setpoint determination methodology.

7.2.3.2 *Evaluation of Reactor Protection System Conformance with Acceptance Criteria – Other Criteria*

SRP Section 7.2 states that RPS design should be evaluated for conformance to IEEE Std 603. This section evaluates conformance with IEEE Std 603 criteria not previously evaluated in Section 7.2.3.1 of this report.

The staff evaluated conformance with IEEE Std 603, Sections 5.2, 5.9, 5.10, 5.11, 5.12, 6.1, 6.2. 6.4, 6.6, 6.7, 6.8, 7.1, 7.2, 7.3, 7.4, 7.5, 8.1, 8.2, and 8.3. Section 7.1.1.3.10 of this report evaluates these criteria, and the staff finds that these criteria are adequately addressed based on their inclusion in the safety system design basis and their verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the RPS. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for these criteria apply to the RPS. Accordingly, the staff finds that conformance to IEEE Std 603, Sections 5.2, 5.9, 5.10, 5.11, 5.12, 6.1, 6.2. 6.4, 6.6, 6.7, 6.8, 7.1, 7.2, 7.3, 7.4, 7.5, 8.1, 8.2, and 8.3 are adequately addressed for the RPS.

The staff evaluated conformance with IEEE Std 603, Section 5.4. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 5.4 is adequately addressed based on its inclusion in the safety system design basis and verification of the EQ in the DCD Tier 1, Revision 9, Section 3.8, ITAAC. This evaluation applies to the RPS. Accordingly, the staff finds that conformance to IEEE Std 603, Section 5.4 is adequately addressed for the RPS.

The staff evaluated conformance with IEEE Std 603, Section 5.5. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 5.5 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, Section 3.2, DAC/ITAAC, and Section 3.8, ITAAC. This evaluation applies to the RPS. Accordingly, the staff finds that conformance to IEEE Std 603, Section 5.5 is adequately addressed for the RPS.

The staff evaluated conformance with IEEE Std 603, Section 5.13. Section 7.1.1.3.10 of this report evaluates this criterion. The multi-unit station criteria do not apply to the standard single unit plant design submitted for NRC certification. The staff finds that IEEE Std 603, Section 5.13, is not applicable to design certification.

The staff evaluated conformance with IEEE Std 603, Section 5.14. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 5.14 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 3.3, DAC/ITAAC. This evaluation applies to the RPS. Accordingly, the staff finds that conformance to IEEE Std 603, Section 5.14 is adequately addressed.

The staff evaluated conformance with IEEE Std 603, Section 5.15. Section 7.1.1.3.10 of this report evaluates this criterion, and the staff finds that Section 5.15 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15 DAC/ITAAC for IEEE Std 603, Section 5.1; DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC; and DCD Tier 1, Revision 9, Section 3.6, ITAAC. This evaluation applies to the RPS. Accordingly, the staff finds that Section 5.15 is adequately addressed for the RPS.

7.2.3.3 *Evaluation of Reactor Protection System Compliance with Regulations and Conformance to the Staff Requirements Memorandum on SECY-93-087*

The staff reviewed the regulations in the acceptance criteria for the RPS in accordance with SRP Section 7.2 and SRP Appendix 7.1-A. For several of the GDC, compliance can be satisfied by meeting IEEE Std 603 requirements, which the staff evaluated in the previous two sections. Compliance with IEEE Std 603 is briefly discussed with the relevant GDC, including the use of DAC, consistent with both SRP sections.

GDC 1 requires quality standards and maintenance of appropriate records. 10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The staff evaluated whether GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed for the RPS in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. DCD Tier 2, Revision 9, Table 7.1-1 identifies that GDC 1 and 10 CFR 50.55a(a)(1) apply to the RPS. The staff evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1) and GDC 1 in Section 7.1.1.3.3 and 7.1.1.3.6 of this report applies to the RPS. Accordingly, because the applicant identified relevant codes and standards applicable to the RPS, the staff finds that the requirements of GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the RPS. SRP Section 7.2 identifies that GDC 2 and 4 are addressed by identification of those systems and components for the RPS designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles in the design bases. SRP Section 7.2 also identifies that GDC 2 and 4 are addressed by the review of the qualification program in DCD Tier 2, Revision 9, Sections 3.10 and 3.11. DCD Tier 2, Revision 9, Table 7.1-1 identifies that GDC 2 and 4 apply to the RPS. DCD Tier 2, Revision 9, Table 3.2-1, identifies that the safety RPS are designed as seismic Category I systems. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Chapter 3 of this report. DCD Tier 1, Revision 9, Table 3.8-1, Items 1 and 3, include ITAAC for the applicant to verify the EQ of safety electrical and digital I&C equipment. The evaluation of GDC 2 and 4 in Section 7.1.1.3.6 of this report further addresses these topics and applies to the RPS. Accordingly, based on the applicant's identification of EQ programs consistent with the design bases for the RPS and their verification in the DCD Tier 1, Revision 9, ITAAC, the staff finds that the requirements of GDC 2 and 4 are adequately addressed.

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection system be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 10 and 15 are adequately addressed for the RPS. SRP Appendix 7.1-A states, for GDC 10 and GDC 15, that the staff review should evaluate the I&C system contributions to design margin for reactor core and reactor coolant systems. DCD Tier 2, Revision 9, Table 15.1-6, identifies systems, including the RPS, required to mitigate AOOs affecting the reactor core and reactor coolant systems. DCD Tier 2, Revision 9, Chapter 7, includes corresponding actions in the design bases of the RPS to maintain the reactor core and reactor coolant system within appropriate margins. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the RPS design implements these design bases. Accordingly, based on the applicant's identification of necessary protection and safety actuations in the design bases for the RPS and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 10 and 15 are adequately addressed.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 13 and 19 are adequately addressed for the RPS. Section 7.1.1.3.6 of this report provides the evaluation of conformance with GDC 19 with the exception of RPS support functions necessary for operating the reactor. SRP Section 7.2 identifies that GDC 13 and 19 are addressed by the review of status information and manual initiation capabilities. DCD Tier 2, Revision 9, Section 7.2.1.2.4.3, specifies that the MCR displays provide status information and alarms for RPS related variables. DCD Tier 2, Revision 9, Section 7.2.1.5.2 specifies the automatic and manual bypass of selected scram functions for the RPS. DCD Tier 2, Revision 9, Section 7.2.1.2.4.2, specifies the manual scram capabilities of the RPS. DCD Tier 2, Revision 9, Section 7.2.1.5.3, specifies the RPS manual controls and Section 7.2.1.5.4 specifies the features of the reactor mode switch. DCD Tier 2, Revision 9, Section 7.2.2.5, specifies the NMS displays and alarms. In combination with the following identified interrelated processes to design the monitoring capability and controls for the RPS, the staff finds these monitoring capabilities and controls acceptable.

NEDE-33226P and NEDE-33245P, as part of a software life cycle process, define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. DCD Tier 1, Revision 9, Section 3.3, includes the

DAC/ITAAC for implementing an HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. These verifications apply to the RPS and include verifying the controls for manual initiation and control of RPS functions necessary to support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Accordingly, based on identified RPS monitoring and controls capabilities, the defined processes for completing the design and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed.

GDC 20 requires that the protection system be designed to (1) initiate automatically the operation of the appropriate systems, including reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs and (2) sense accident conditions and initiate the operation of systems and components important to safety. The staff evaluated whether GDC 20 is adequately addressed for the RPS. SRP Appendix 7.1-A notes that GDC 20 is addressed for protection systems by conformance with IEEE Std 603, Sections 4, 5, 5.5, 6.1, 6.8, and 7.1. The applicant has committed to following the guidance of RG 1.105 and has provided a setpoint methodology in NEDE-33304P. The staff also evaluated in Section 7.2.3.1 of this report the RPS design-basis requirements, general functional requirements, and system integrity, which involve IEEE Std 603, Sections 4, 5, 5.5, 6.1, 6.8, and 7.1, in Section 7.2.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the safety functions of setpoints are defined, determined, and implemented based on the defined setpoint methodology and that the design is completed in compliance with IEEE Std 603. DCD Tier 1, Revision 9, Table 2.2.15-2, also includes DAC/ITAAC for applicable sections of IEEE Std 603, Section 4, for verifying that complete design basis information is identified and implemented for RPS related software projects. The staff's evaluation of conformance to IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the RPS. Accordingly, based on the applicant's identification of design bases for the RPS, conformance to applicable guidance and IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 20 are adequately addressed.

GDC 21 requires that protection systems be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. The staff evaluated whether GDC 21 is adequately addressed for the RPS. SRP Appendix 7.1-A states that GDC 21 is addressed for protections systems by conformance to IEEE Std 603 criteria, except for Sections 5.4, 6.1, and 7.1. In addition, SRP Section 7.2 identifies that GDC 21 is addressed by conformance to RGs 1.22, 1.47, 1.53, and 1.118 and IEEE Std 379. DCD Tier 2, Revision 9, Table 7.1-1, identifies that the guidelines for periodic testing in RGs 1.22 and 1.118, apply to the RPS. The bypassed and inoperable status indication conforms to the guidelines of RG 1.47. DCD Tier 2, Revision 9, Section 7.1.2.4, states that the DCIS conforms to the guidelines on the application for the single failure criterion in IEEE Std 379, as supplemented by RG 1.53. DCD Tier 1, Revision 9, Section 7.1.6.6.1.2, states that FMEAs complying with IEEE Std 379 will be used to confirm the safety-related systems designs' conformance to the IEEE Std 603, Section 5.1. DCD Tier 2, Revision 9, Section 7.2, describes the conformance of RPS to IEEE Std 603, which is evaluated in Section 7.2.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the RPS design is completed in compliance with IEEE Std 603. In particular, DCD Tier 1, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Sections 5.1, 5.7 and 6.5. DCD Tier 1, Table 2.2.15-2, also includes DAC/ITAAC for applicable sections of IEEE Std 603, Section 4, for verifying that complete design basis information is identified and implemented for RPS related software projects. Accordingly, based on the applicant's identification of design bases for the RPS, conformance to applicable guidance and

IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 21 are adequately addressed.

GDC 22 requires, in the pertinent part, that protection systems be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels, do not result in loss of the protection function. The staff evaluated whether GDC 22 is adequately addressed for the RPS. SRP Appendix 7.1-A states that GDC 22 is addressed for protection systems by conformance to IEEE Std 603, Sections 4, 5.1, 5.3, 5.4, 5.5, 5.6, 6.2, 6.3, 6.8, 7.2, and 8. In addition, SRP Section 7.2 identifies that GDC 22 is addressed by conformance to RG 1.75. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 22 and RG 1.75 apply to the RPS. DCD Tier 2, Revision 9, Section 7.2, describes the conformance of the RPS to IEEE Std 603, which is evaluated in Section 7.2.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the RPS complies with IEEE Std 603. In particular, DCD Tier 1, Table 2.2.15-2, provides the DAC/ITAAC for IEEE Std 603, Section 5.6. DCD Tier 1, Table 2.2.15-2, also includes the DAC/ITAAC for applicable sections of IEEE Std 603, Section 4, for verifying that complete design basis information is identified and implemented for RPS related software projects. Accordingly, based on the applicant's identification of design bases for the RPS, conformance to applicable guidance and IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 22 are adequately addressed.

GDC 23 requires that protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if certain conditions are experienced. The staff evaluated whether GDC 23 is adequately addressed for the RPS. SRP Appendix 7.1-A notes that GDC 23 is addressed for protection systems by conformance to IEEE Std 603, Section 5.5. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 23 applies to the RPS. DCD Tier 2, Revision 9, Section 7.1.6.6.1.6, states that the RPS fails to a tripped state. Hardware and software failures detected by self-diagnostics cause a trip signal to be generated in the RPS division in which the failure occurs. DCD Tier 1, Revision 9, Table 2.2.15-2 and Sections 3.2 and 3.8 provide ITAAC for verifying that the RPS design is completed in compliance with IEEE Std 603, Section 5.5. Accordingly, based on conformance to the applicable guidance and IEEE Std 603 sections and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 23 are adequately addressed.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for the RPS. SRP Appendix 7.1-A identifies that GDC 24 is addressed for protection systems by conformance to IEEE Std 603, Sections 5.1, 5.6, 5.12, 6.3, 6.6, and 8, particularly Sections 5.6 and 6.3. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 24 applies to the RPS. DCD Tier 2, Revision 9, Section 7.2, describes the conformance of the RPS to IEEE Std 603, which is evaluated in Sections 7.2.3.1 and 7.2.3.2 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the applicable I&C systems' design is completed in compliance with IEEE Std 603, including Sections 5.6 and 6.3. Accordingly, based on the applicant's identification of design bases for the RPS, conformance to applicable IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 24 are adequately addressed.

GDC 25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. The staff evaluated whether GDC 25 is adequately addressed for the RPS. SRP Appendix 7.1-A states that GDC 25 is addressed for protection systems by conformance to IEEE Std 603, Section 4, which is associated with safety system design-basis requirements. DCD Tier 2, Revision 9, Section 7.2.1, provides design bases for the RPS that include protection from AOOs such as continuous control rod withdrawal. DCD Tier 2, Revision 9, Chapter 15, includes analysis for continuous rod withdrawal in several scenarios; the RPS is designed to prevent fuel design limits from being exceeded. DCD Tier 1, Revision 9, Tables 2.2.1-6 and 2.2.7-4, provide ITAAC for verification of these reactor protection functions. DCD Tier 1, Revision 9, Table 2.2.15-2 includes the DAC/ITAAC for applicable sections of IEEE Std 603, Section 4, for verifying that complete design basis information is identified and implemented for RPS related software projects. Accordingly, based on the applicant's identification of design bases for the RPS, conformance to applicable IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 25 are adequately addressed.

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed for the RPS. SRP Appendix 7.1-A notes that GDC 29 is addressed by conformance as applicable to GDC 20–25 and GDC 28. However, GDC 28, which applies to reactivity control systems, is not applicable to the protection systems evaluated in this section. Accordingly, GDC 29 is addressed by conformance, as applicable, to GDC 20–25. DCD Tier 2, Revision 9, Table 7.1-1 and Section 7.2, indicate that applicable sub-systems of the RPS conform to GDC 29. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the applicable RPS designs implement these design criteria. Accordingly, based on the applicant's identification of design bases for the RPS, conformance to applicable guidance and IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, ITAAC, the staff finds that the requirements of GDC 29 are adequately addressed.

Section 7.1.1.3.7 of this report documents the staff evaluation of the I&C system in response to the SRM to SECY-93-087. This evaluation applies to the RPS. Accordingly, the staff finds that the guidelines of SRM to SECY-93-087 are adequately addressed for the RPS.

The staff evaluated whether 10 CFR 50.34(f)(2)(v) and 10 CFR 50.34(f)(2)(xxiii) are adequately addressed for the RPS. As described in Section 7.1.1.3.4 of this report, the staff evaluated the I&C system design's compliance with 10 CFR 50.34(f)(2)(v) and 10 CFR 50.34(f)(2)(xxiii) and finds it acceptable. This evaluation applies to the RPS. Accordingly the staff finds that the requirements of 10 CFR 50.34(f)(2)(v) and 10 CFR 50.34(f)(2)(xxiii) are adequately addressed for the RPS.

The staff evaluated whether 10 CFR 50.55a(h) is adequately addressed for the RPS. The regulation at 10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995, which the staff evaluated in Sections 7.2.3.1 and 7.2.3.2 of this report and finds to be adequately addressed. Accordingly, the staff finds that 10 CFR 50.55a(h) is adequately addressed for the RPS.

The staff evaluated whether the applicant met the requirements of 10 CFR 52.47(b)(1). This regulation requires that the application for design certification contain 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. Section 7.2.3 of this report addresses the ITAAC specific to the RPS. The staff evaluation of conformance to 10 CFR 52.47 in Section 7.1.1.3.4 of this report applies to the RPS. Therefore, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the RPS.

7.2.4 Conclusion

Based on the above, the staff concludes that the applicant adequately addresses the major design considerations for the RPS. As discussed in Sections 7.1.1.3.1 through 7.1.1.3.10 of this report and Section 7.2.3 above, the staff concludes that, for the RPS, the applicant adequately addresses the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f); 10 CFR 52.47(b)(1); GDC 1, 2, 4, 10, 13, 15, 19 – 25 and 29; and the guidelines of the SRM to SECY-93-087. The applicant also identifies adequate high-level functions and includes sufficient DAC/ITAAC in DCD Tier 1 to verify that the RPS design is completed in compliance with the applicable requirements.

7.3 Engineered Safety Features Systems

7.3.1 Regulatory Criteria

The objective of the review of DCD Tier 1, Section 2.2, and DCD Tier 2, Section 7.3, is to confirm that the ESF actuation and control systems, VBIF, and all subsystems satisfy regulatory acceptance criteria, guidelines, and performance requirements. The review of I&C aspects of the ESF systems includes the ESF actuation and control systems, VBIF, and all subsystems. The ESF actuation systems detect a plant condition requiring the operation of an ESF system, auxiliary supporting features, or both, and other auxiliary features and initiate operation of the systems. The ESF control systems regulate the operation of the ESF systems following automatic initiation by the protection system or manual initiation by the plant operator. In the design, the SSLC/ESF system performs the ESF actuation and control systems, VBIF, and all subsystem functions.

Acceptance criteria in SRP Section 7.3, Revision 5, for the ESF actuation and control systems, VBIF, and all subsystems, and hence the SSLC/ESF system, are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(v), 10 CFR 50.34(f)(2)(xii), and 10 CFR 50.34(f)(2)(xiv); 10 CFR 52.47(b)(1); and GDC 1, 2, 4, 10, 13, 15, 16, 19, 20, 21, 22, 23, 24, 29, 33, 34, 35, 38, 41, and 44. The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152 and the SRM to SECY-93-087.

7.3.2 Summary of Technical Information

7.3.2.1 *Engineered Safety Features Systems Description*

The I&C ESF systems are part of a group of systems that are collectively referred to as the Q-DCIS. DCD Tier 2, Section 7.3, describes the ESF systems. The ESF systems for the design include the following:

- ECCS
- PCCS
- LD&IS
- CRHS
- SSLC/ESF system
- Vacuum Breaker Isolation Function (VBIF)
- ICS DPV Isolation Function

7.3.2.1.1 Emergency Core Cooling System

The ECCS comprises the ADS, GDCS, ICS, and the SLC system. Section 7.4 of this report evaluates the ICS and the SLC system.

The ADS resides within the NBS. It depressurizes the reactor so that the low-pressure GDCS can provide makeup coolant to the RPV. The ADS I&C perform the following safety functions:

- Detect reactor low water level, Level 1.

- Automatically actuate the SRVs and DPVs after Level 1 is reached or drywell pressure high is detected.

- Actuate the SRVs and DPVs sequentially and in groups to achieve the required depressurization characteristics.

- Indicate the status of SRVs and DPVs in the MCR.

The GDCS I&C perform the following functions:

- Automatically initiate the GDCS to prevent fuel cladding temperatures from reaching the limits of 10 CFR 50.46.

- Respond to a need for emergency core cooling, following reactor depressurization, regardless of the physical location of the malfunction or break that causes the need.

- Be completely automatic in operation (that is, no operator action required). Manual initiation of GDCS is possible at any time, providing that protective interlocks are satisfied (e.g., the reactor is depressurized).

- Prevent the inadvertent actuation of the deluge valves thus preventing inadvertent draining of the GDCS pools.

- Prevent any single control logic and instrumentation failure from inadvertently opening a GDCS injection valve or equalizing valve.

- Display GDCS valve positions and GDCS pool levels on the mimic on the WDP in the MCR.

7.3.2.1.2 Passive Containment Cooling System

The PCCS consists of condensers that are an integral part of the containment pressure boundary. The PCCS heat exchanger tubes are located in the isolation condenser/passive containment cooling system (IC/PCCS) pool of water outside the containment. A rise in containment (drywell) pressure above the suppression pool (wetwell) pressure, similar to the situation during a loss of reactor coolant into the drywell, forces flow through the PCCS condensers. Condensate from the PCCS drains to the GDCS pools. As the flow passes through the PCCS condensers, heat is rejected to the IC/PCCS pool, thereby cooling the containment atmosphere. This action occurs automatically, without the need for actuation of components. The PCCS does not have instrumentation, control logic, or power-actuated valves and does not need or use electrical power for its operation in the first 72 hours after a LOCA. While the PCCS has no I&C functions, it does rely on I&C functions in other systems, namely the ICS, SSLC/ESF, and FAPCS to perform its safety functions. The PCCS relies on the water in the Equipment Storage Pool and Reactor Well to perform its safety functions for 72 hours. Pool cross-connect valves are active safety components that open to allow water in the Equipment Storage Pool and Reactor Well to flow into the IC/PCCS pools. The FAPCS provides four safety-related level sensors in each IC/PCCS inner expansion pool. The cross-connect valves are opened when the sensors in either pool detect a low level condition. The FAPCS also provides four nonsafety level sensors in each inner expansion pool which are used by the DPS to open the cross-connect valves. The air-operated cross-connect valves require pneumatic and electrical motive power to open, which is provided by a pneumatic accumulator, and the safety-related UPS. The squib cross-connect valves are opened pyrotechnically and need only electrical motive power to open, which is provided by the safety-related UPS. For long-term effectiveness of the PCCS, the vent fans and their isolation valves are automatically or manually initiated. For severe accident events, igniters were added to the lower drum of each PCCS heat exchanger to prevent the accumulation of explosive mixtures of hydrogen and oxygen with simultaneous containment high pressure conditions.

7.3.2.1.3 Leak Detection and Isolation System

The primary function of the LD&IS is to detect and monitor leakage from the RCPB and to initiate the appropriate safety action to isolate the source of the leak. The system is designed to automatically initiate the isolation of certain designated process lines penetrating the containment to prevent release of radioactive material from the RCPB. The initiation of the isolation functions closes the appropriate containment isolation valves. The LD&IS functions are performed in two separate and diverse safety platforms. The MSIV isolation logic functions are performed in the RTIF platform (evaluated in Section 7.2 of this report), while all other containment isolation logic functions are performed in the SSLC/ESF platform. The containment isolation function of LD&IS logic design is fail as-is, such that a loss of power to the logic of one division does not result in a trip. The LD&IS logic design is fail-safe, such that a loss of electrical power to one LD&IS divisional logic channel initiates a channel trip. The LD&IS control and isolation logic uses 2/4 coincidence voting channels for each plant variable monitored for containment isolation. Various plant variables are monitored, such as flow, temperature, pressure, RPV water level, and radiation level. These are used in the logic to initiate alarms and the required control signals for containment isolation. Two or more diverse leakage parameters are monitored for each specific isolation function. The LD&IS logic functions reside in the framework of the RTIF and the SSLC/ESF platforms, where trip signals are generated, initiating the isolation functions of the LD&IS. This system operates continuously during normal reactor operation, and during abnormal and accident plant conditions.

7.3.2.1.4 Main Control Room Habitability System

The CRHS is an ESF system that provides a safe environment within the MCR, allowing the operator or operators to do the following:

- Control the nuclear reactor and its auxiliary systems during normal conditions
- Safely shut down the reactor
- Maintain the reactor in a safe condition during abnormal events and accidents

The CRHS safety I&C (part of the SSLC/ESF platform) are designed to isolate the MCR envelope upon detection of the following signals and realign to the emergency filtration mode:

- High radiation in the inlet air supply (automatic action – safety function)

- Loss of ac power, SBO, (automatic action – safety function)

- Smoke in the inlet air supply or smoke in the CRHS general area (manual isolation – nonsafety function)

Additional CRHS safety instrumentation is designed to only swap over the operating emergency filtration train after the following situations:

- Detection of high radiation downstream of the operating emergency filter unit (EFU) filter train (automatic action) (nonsafety function)

- Detection of low flow at the outlet of the operating EFU filter train (automatic action) (safety function)

7.3.2.1.5 Safety System Logic and Control/Engineered Safety Features System

The SSLC/ESF system processes automatic and manual demands for ESF system actuations based on sensed plant process parameters or operator request. The SSLC/ESF runs without interruption in all modes of plant operation to support the required safety functions. The SSLC/ESF system includes the controls and instruments that implement the non-MSIV isolation functions of the LD&IS; CRHS; the ECCS functions that include the ADS, GDCS, and the SLC system; and the ECCS and shutdown functions of the ICS.

The SSLC/ESF platform provides the following functions:

- Monitor safety signals that provide automatic control of the plant safety protection systems.

- Perform processing of plant sensor and equipment interlock logic signals according to the required trip and interlock logic, including time delays, of each safety interfacing plant system or system important to safe plant operation.

- Meet the performance requirements of each safety interfacing plant system or system important to safe plant operation, including transient response, delay time, and overall time to trip system actuators or initiate necessary system operation.

- Monitor safety manual control switches used for system or component test, protection system manual initiation, and individual control of equipment actuators.

- Furnish trip outputs signals to actuators driving safety system equipment (e.g., solenoids and squib explosive-actuated valves).

- Furnish trip or initiation output signals to the logic of interfacing functions.

- Provide diagnostic capabilities for detecting failure of safety system components and provide an operator interface that facilitates quick repair.

- Provide safety accident monitoring display information, alarm, and status outputs to operator displays, annunciators and the plant computer.

7.3.2.1.6 Containment System Wetwell-to-Drywell Vacuum Breaker Isolation Function

The VBIF is an independent control platform that, upon detection of excessive vacuum breaker (VB) leakage, prevents the loss of long-term containment integrity. The wetwell-to-drywell VB isolation function has the following safety requirements:

- The function automatically isolates an excessively leaking VB using a VB isolation valve.

- The VB and VB isolation valve are qualified for a harsh environment inside the drywell.

- Manual opening and closing of a VB isolation valve are provided.

- No single control logic and instrument failure will open and close more than one VB isolation valve.

- VB and VB isolation valve positions are displayed in the MCR.

- The safety function is met with one VB/VB isolation valve path isolated together with any active identifiable single failure.

- Divisional instruments performing VB isolation valve logic are powered by the associated safety divisional power supplies.

- VB isolation function logic controllers use a platform that is independent and diverse from the RTIF-NMS and the SSLC/ESF platforms.

- Containment system VBIF logic controllers are independent.

7.3.2.1.7 ICS DPV Isolation Function

The ICS DPV isolation function which is implemented in the ICP prevents the loss of long-term containment integrity upon detection of DPV open position.

The ICS DPV isolation function has the following safety-related requirements:

- Automatically isolates all Isolation Condensers by closing the two steam admission isolation valves to each of the ICs.

- The two steam admission isolation valves per IC are qualified for a harsh environment inside the drywell.

- Manual opening and closing of the IC steam admission isolation valves is provided for in the design.

- No single control logic and instrumentation failure opens/closes more than one IC steam admission isolation valve.

- IC steam admission isolation valve positions are displayed in the MCR.

- The safety-related function is met with one IC steam admission valve path isolated together with any active identifiable single failure.

- Divisional instruments performing IC steam admission valve isolation valve logic are powered by the associated safety-related divisional power supplies.

- ICS DPV isolation function logic controllers are independent.

7.3.2.2 *Safety System Logic and Control/Engineered Safety Features System Architecture*

The SSLC/ESF resides in four independent and separated instrumentation divisions. The SSLC/ESF integrates the control logic of the safety systems in each division into microprocessor-based, software-controlled processing modules located in divisional cabinets in the safety equipment room of the control building. Most SSLC/ESF input data are process variables multiplexed via the Q-DCIS in four physically and electrically isolated redundant instrumentation divisions. Each of the four independent and separated Q-DCIS divisions uses triply redundant processors to implement all divisional logic. All input data are processed within the RMUs function of the Q-DCIS. The sensor data are transmitted through the triply redundant SSLC/ESF specific network to the SSLC/ESF system's triply redundant DTM function for setpoint comparison. A trip (or actuation) signal is generated from this function. Processed trip signals from a division and trip signals from the other three divisions are transmitted through the CIMs and are processed in the triply redundant voter logic unit (VLU) function for 2/4 voting. The VLU trip outputs are sent via triply redundant paths to the RMUs which have three series connected load drivers (for DPS valves) or two series connected load drivers (for solenoid operated SRVs). Each of the load drivers uses two-out-of-three (2/3) voting to close its output contacts. The redundant processors and series connected load drivers within a division are necessary to prevent single failures within a division from causing a squib initiator to fire; all series load drivers within a division must operate to get an output. Self-tests within the SSLC/ESF determine if any one VLU function has failed, and the failure is alarmed in the MCR. To prevent single I&C failure from causing inadvertent actuations, a failed VLU function cannot be bypassed for any of the ECCS logic for squib valves initiation. Trip signals are hard-wired from the RMUs to the equipment actuator. The final trip signal (from two or more divisions) is then transmitted to the RMUs function via the Q-DCIS network to initiate mechanical actuation devices.

At the division level, the four redundant divisions provide a fault-tolerant architecture that allows single division of sensor bypass for online maintenance, testing, and repair with the intent of not losing trip capability. In bypass condition (i.e., when a division of sensor inputs is bypassed), the system automatically defaults to 2/3 coincident voting.

7.3.3 Staff Evaluation

The staff reviewed the SSLC/ESF system in accordance with SRP Section 7.3. The staff also used acceptance criteria in SRP Section 7.1, SRP Table 7-1, SRP Appendix 7.1-A, and SRP

Appendix 7.1-C, as directed by SRP Section 7.3. Section 7.3.1 of this report describes the acceptance criteria listing used as the basis for the staff review of the SSLC/ESF system.

SRP Section 7.1 describes the procedures to be followed in reviewing any I&C system. SRP Section 7.3 highlights specific topics that should be emphasized in reviewing the ESF actuation and control systems, VBIF, and all subsystems; Section 7.3.3.1 of this report addresses these topics. The staff included the review of the DCD Tier 1, Revision 9, DAC/ITAAC during the review of this section because of their significant role in determining the SSLC/ESF system conformance to all requirements.

As described in Section 7.1.1.3.1 of this report, the DCD does not provide the required SSLC/ESF system design information to comply with IEEE Std 603. Instead, the applicant included the DAC/ITAAC in DCD Tier 1, Revision 9, Section 2.2.15, to confirm that the completed SSLC/ESF system design complies with IEEE Std 603. DCD Tier 1, Section 2.2.15, also includes an ITAAC applicability table (Table 2.2.15-1), which identifies the applicability of the IEEE Std 603 criteria DAC/ITAAC to the SSLC/ESF system. The staff has accepted the DAC approach to addressing compliance with IEEE Std 603. The staff evaluation of conformance to IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the SSLC/ESF.

In RAI 7.1-99, the staff asked the applicant to clarify the applicability of IEEE Std 603 criteria in a consistent manner throughout DCD Tier 2, Chapter 7. The applicant submitted a response to RAI 7.1-99, along with responses to RAIs 7.1-100, 7.1-101, and RAI 14.3-265 S01, all of which are incorporated in DCD, Revision 8. Section 7.1.1.3.1 of this report provides additional discussion on the resolution of these RAIs. With regard to the staff's evaluation of the SSLC/ESF system, the applicant significantly revised DCD Tier 2, Tables 7.1-1 and 7.1-2, to clearly identify the applicability of IEEE Std 603 criteria to each SSLC/ESF subsystem. Concurrently, the applicant revised or removed many references to IEEE Std 603 criteria from the discussions of SSLC/ESF subsystems in DCD Tier 2, Section 7.3, to address the staff's consistency concerns. Accordingly, the staff has updated or removed statements regarding the applicability of IEEE Std 603 criteria found in the SER with open items from the remainder of Section 7.3.3 of this report to be consistent with DCD Revision 8.

7.3.3.1 *Evaluation of Engineered Safety Features Actuation and Control Systems Conformance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.3, the following are the major design considerations that should be emphasized in the staff's review of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(1) Design Basis (IEEE Std 603, Section 4)

The staff evaluated the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems design basis to determine whether IEEE Std 603, Section 4, is adequately addressed using SRP Appendix 7.1-C, Section 4, "Safety System Designation (IEEE Std 603)." For completeness, the SRP states, "As a minimum each of the safety system design basis aspects identified in IEEE Std 603, Sections 4.1 through 4.12 should be addressed." Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 4 is adequately addressed based on its inclusion in the safety system design basis and the verification of applicable criteria in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the SSLC/ESF system. In addition, DCD Tier 1, Revision 9,

Table 2.2.15-1, identifies that the DAC/ITAAC for applicable criteria of Section 4 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

DCD Tier 2, Revision 9, Section 7.3, identifies individual parameters that determine operation of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. As mentioned previously, NEDE-33226P and NEDE-33245P, as part of the software life cycle process, define a process by which plant performance requirements, including response times, under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. Accordingly, the staff finds that Section 4 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(2) Single Failure Criterion (IEEE Std 603, Section 5.1)

The staff evaluated whether the single failure criterion, IEEE Std 603, Section 5.1, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. This criterion requires that any single failure within the safety system shall not prevent proper protective action at the system level when required. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.1 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 5.1 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that Section 5.1 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(3) Quality of Components and Modules (IEEE Std 603, Section 5.3)

The staff evaluated whether the quality criterion, IEEE Std 603, Section 5.3, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.3 is adequately addressed based on the inclusion of IEEE Std 7-4.3.2, Section 5.3, in the safety system design basis and the verification of the software development activities in the DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC. In addition, the applicant stated that the quality assurance program conforms to GDC 1. Chapter 17 of this report addresses the evaluation of the adequacy of the quality assurance program. These evaluations apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.4, also discusses the applicability of this criterion to the Q-DCIS design. Accordingly, based on the applicant’s use of an acceptable software development process, as evaluated in Section 7.1.2.3 of this report, and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that 5.3 of IEEE Std 7-4.3.2 and Section 5.3 of IEEE Std 603 are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(4) Independence (IEEE Std 603, Sections 5.6 and 6.3)

The staff evaluated whether the independence-related criteria, IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, Section 5.6. The staff finds that Section 5.6 is adequately addressed based on its inclusion in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. In addition DCD Tier 1, Revision 9, Table 2.2.15-1 identifies that the DAC/ITAAC for Section 5.6 applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.6, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated conformance with IEEE Std 603, Section 6.3. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 6.3 is adequately addressed based on its inclusion in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1 identifies that the DAC/ITAAC for Section 6.3 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 6.3, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(5) Completion of Protective Action (IEEE Std 603, Section 5.2)

The staff evaluated whether the completion of the protective action criterion, IEEE Std 603, Section 5.2, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.2 is adequately addressed based on its inclusion in the safety system design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that IEEE Std 603, Section 5.2, applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.2, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(6) Diversity and Defense-in-Depth

The staff evaluated whether the D3 criteria are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.3 of this report provides a general evaluation of the conformance of safety I&C systems to the D3 criteria. For ESF systems, these criteria identify that the systems should incorporate multiple means for responding to each event discussed in DCD Tier 2, Revision 9, Chapter 15. At least one pair of these means for each event should have the property of signal diversity. In the ESBWR design, the applicant has implemented the diversity and defense-in-depth principle (i.e., the use of different sensed parameters to initiate protective action), in which any of the parameters may independently indicate an abnormal condition even if the other parameters are sensed incorrectly (see NUREG/CR–6303). NEDO-33251 asserts conformity to NUREG/CR–6303. DCD Tier 1, Revision 9, Table 2.2.14-4, includes ITAAC for the applicant to complete an FMEA of the safety protection system platforms to validate the DPS functions, in accordance with NUREG/CR–6303. The staff's evaluation of conformance to D3 in Section 7.1.3.3 of this report applies to the SSLC/ESF system and the VB isolation function. Accordingly, the staff finds that D3 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(7) System Testing and Inoperable Surveillance (IEEE Std 603, Sections 5.7, 5.8, and 6.5)

The staff evaluated whether the criteria related to system testing and inoperable surveillance, IEEE Std 603, Sections 5.7, 5.8, and 6.5, are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, Section 5.7. The staff finds that Section 5.7 is adequately addressed based on its inclusion in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for Section 5.7 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.7, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated whether IEEE Std 603, Section 5.8, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.8 is adequately addressed based on its inclusion in the safety systems design basis and its verification in the DCD Tier 1, Revision 9, Section 3.3, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 2, Revision 9, Chapter 18, describes the HFE design process to design information displays, which is evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification, which applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, includes the inventory of displays for manually controlled actions, system status indications, and indications of bypasses. Accordingly, the staff finds that IEEE Std 603, Section 5.8, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated conformance with IEEE Std 603, Section 6.5. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 6.5 is adequately addressed based on its inclusion in the safety systems’ design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 1, Revision 9, Table 2.2.15-1 identifies that the DAC/ITAAC for Section 6.5 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 6.5, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(8) Use of Digital Systems (IEEE Std 7-4.3.2)

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2 concerning the use of digital systems. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in SRP Appendix 7.1-D. The staff evaluation of conformance to IEEE Std 7-4.3.2 in Section 7.1.1.3.10 of this report applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

NEDE-33226P describes the software development activities and NEDE-33245P describes the software QA activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff evaluation of software development and software QA activities. Accordingly, the staff finds that the use of digital systems is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

(9) Setpoint Determination

Section 7.1.4 of this report evaluates the setpoint determination methodology.

7.3.3.2 *Evaluation of the Conformance of Engineered Safety Features Actuation and Control Systems with Acceptance Criteria - Other Criteria*

SRP Section 7.3 states that the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems designs should be evaluated for conformance to IEEE Std 603. This section evaluates conformance with IEEE Std 603 criteria not previously evaluated in Section 7.3.3.1 of this report.

The staff evaluated conformance with IEEE Std 603, Sections 5.9, 5.10, 5.11, 5.12, 6.1, 6.2. 6.4, 6.6, 6.7, 6.8, 7.1, 7.2, 7.3, 7.4, 7.5, 8.1, 8.2, and 8.3. Section 7.1.1.3.10 of this report evaluates these conformance with criteria, and the staff finds that these criteria are adequately addressed based on their inclusion in the safety systems' design basis and their verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for these criteria apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that conformance to IEEE Std 603, Sections 5.9, 5.10, 5.11, 5.12, 6.1, 6.2. 6.4, 6.6, 6.7, 6.8, 7.1, 7.2, 7.3, 7.4, 7.5, 8.1, 8.2, and 8.3 are adequately addressed for the applicable to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated conformance to IEEE Std 603, Section 5.4. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.4 is adequately addressed based on its inclusion in the safety systems' design basis and verification of the EQ in the DCD Tier 1, Revision 9, Section 3.8, ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.4, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated conformance to IEEE Std 603, Section 5.5. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.5 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, Section 3.2, DAC/ITAAC, and Section 3.8, ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.5, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated conformance with IEEE Std 603, Section 5.13. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The multi-unit station criteria do not apply to the standard single unit plant design submitted for NRC certification. The staff finds that IEEE Std 603, Section 5.13, is not applicable to the ESBWR design certification.

The staff evaluated conformance with IEEE Std 603, Section 5.14. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.14 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 3.3, DAC/ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.14, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated conformance with IEEE Std 603, Section 5.15. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.15 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC for IEEE Std 603, Section 5.1; DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC; and DCD Tier 1, Revision 9, Section 3.6, ITAAC. This evaluation applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that IEEE Std 603, Section 5.15, is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

7.3.3.3 *Evaluation of Engineered Safety Features Actuation and Control Systems Compliance with Regulations and Conformance to the Staff Requirements Memorandum on SECY-93-087*

The staff reviewed the regulations in the acceptance criteria for the ESF actuation and control systems in accordance with SRP Section 7.3 and SRP Appendix 7.1-A. For several of the GDC, compliance can be satisfied by meeting IEEE Std 603 requirements, which the staff evaluated in the previous two sections. Compliance with IEEE Std 603 is briefly discussed below, along with the relevant GDC, including the use of DAC, consistent with both SRP sections.

GDC 1 requires quality standards and maintenance of appropriate records.
10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The staff evaluated whether GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems in accordance with SRP Appendix 7.1-A, which states that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 1 and 10 CFR 50.55a(a)(1) apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. The staff evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1) and GDC 1 in Sections 7.1.1.3.3 and 7.1.1.3.6 of this report applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, based on the identification of codes and standards applicable to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, the staff finds that the requirements of GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Section 7.3 identifies that GDC 2 and 4 are addressed by the identification of those systems and components for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems designed to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles in the design bases. SRP Section 7.3 also identifies that GDC 2 and 4 are addressed by the review of the qualification program in DCD Tier 2, Sections 3.10 and 3.11. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 2 and 4 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 2, Revision 9, Table 3.2-1, identifies that the safety ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems are designed as seismic Category I systems. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Chapter 3 of this report. DCD Tier 1, Revision 9, Table 3.8-1, Items 1 and 3, include ITAAC for the applicant to verify the EQ of safety electrical and digital I&C equipment. The evaluation of GDC 2 and 4 in Section 7.1.1.3.6 of this report further addresses these topics and applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, based on the applicant's identification of EQ programs consistent with the design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, and their verification in the DCD Tier 1, Revision 9, ITAAC, the staff finds that the requirements of GDC 2 and 4 are adequately addressed.

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection system be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 10 and 15 are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Appendix 7.1-A for GDC 10 and 15 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and reactor coolant systems. DCD Tier 2, Revision 9, Table 15.1-6, identifies systems, including ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, required to mitigate AOOs affecting the reactor core and reactor coolant systems. DCD Tier 2, Revision 9, Chapter 7, includes corresponding actions in the design bases of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems to maintain the reactor core and reactor coolant system within appropriate margins. In DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems' design implement these design bases. Accordingly, based on the applicant's identification of necessary protection and safety actuations in the design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 10 and 15 are adequately addressed.

GDC 16 requires that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The staff evaluated whether GDC 16 is adequately addressed for the ESF actuation and control systems. SRP Appendix 7.1-A states that GDC 16 imposes functional requirements on ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems to the extent that they support the requirement that the containment provide a leak tight barrier. DCD Tier 2, Revision 9, Sections 7.3.3 and 7.3.5, identify the containment isolation functions in the LD&IS and ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems design bases. In DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8 include the DAC/ITAAC for the applicant to verify that the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystem designs implement these design bases. Accordingly, based on the applicant's identification of necessary containment isolation functions in the design bases of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 16 are adequately addressed.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 13 and 19 are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Section 7.1.1.3.6 of this report evaluates GDC 19, with the exception of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems support functions necessary for operating the reactor. SRP Section 7.3 identifies that GDC 13 and 19 are addressed by the review of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems status information, manual initiation capabilities, and control capabilities.

DCD Tier 2, Revision 9, Sections 7.3.1.1.2, 7.3.1.2.2, and 7.3.6.2, specify the automatic and manual initiation and control capabilities of the ADS, GDCS, VBIF, and ICS DPVIF, respectively. DCD Tier 2, Revision 9, Sections 7.3.1.1.5, 7.3.1.2.5, and 7.3.6.5, specify the status information and the alarms provided in the MCR for the ADS, GDCS, VBIF, and ICS DPVIF, respectively. DCD Tier 2, Revision 9, Section 7.3.3.2, specifies the automatic controls of the LD&IS. DCD Tier 2, Revision 9, Section 7.3.3.3, specifies the manual initiation and control of the LD&IS. DCD Tier 2, Revision 9, Section 5.2.5.2, specifies the LD&IS monitoring capabilities. DCD Tier 2, Revision 9, Section 5.2.5.2, specifies that (1) monitored plant leakage parameters are measured, recorded and displayed on the appropriate panels in the MCR; (2) all abnormal indications are annunciated for operator alert to initiate corrective action; and (3) all initiated automatic or manual isolation functions are also alarmed in the MCR. DCD Tier 2, Revision 9, Section 7.3.4.2, specifies the automatic and manual initiation and control of the CRHS. DCD Tier 2, Revision 9, Sections 9.4.1.5 and 6.4.8, describe additional control information and status and alarm information. The identified DCD sections above address GDC 13 by identifying the ESF actuation and control systems monitoring and control functions. The identified DCD sections above address GDC 19 by identifying the that the monitoring and control functions are available in the MCR. By design of the Q-DCIS, all monitoring and controls in the MCR are available for remote shutdown.

In combination with the following identified interrelated processes to complete the design of the monitoring capability and controls for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, the staff finds these monitoring capabilities and controls acceptable. NEDE-33226P and NEDE-33245P, as part of a software life cycle process, define processes by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing an HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. These verifications apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems and include verifications of the controls for manual initiation and control of ESF functions necessary to support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Accordingly, based on the identified monitoring capabilities and control room controls, as well as the defined processes for completing their design and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed.

GDC 20 requires that the protection system be designed to (1) initiate automatically the operation of the appropriate systems, including reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs; and (2) sense accident conditions and initiate the operation of systems and components important to safety. The staff evaluated whether GDC 20 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Appendix 7.1-A to the SRP states that GDC 20 is addressed for protection systems by conformance to IEEE Std 603, Sections 4, 5, 5.5, 6.1, 6.8, and 7.1. The applicant has committed to following the guidance of RG 1.105, and has provided a setpoint methodology in NEDE-33304P. Section 7.3.3.1 of this report presents the staff's evaluation of the ESF actuation and control design basis requirements, general functional requirements, and system integrity, which involve IEEE Std 603, Sections 4, 5, 5.5, 6.1, and 7.1. DCD Tier 1, Revision 9, Table 2.2.15-2 includes the DAC/ITAAC for verifying that the safety functions of setpoints are defined, determined, and implemented based on the defined setpoint methodology and for completing the design in compliance with IEEE Std 603. DCD Tier 1, Table 2.2.15-2, also includes DAC/ITAAC for applicable sections of IEEE Std 603, Section 4, for verifying that complete design basis information is identified and implemented for ESF actuation and control systems, VBIF and ICS DPVIF, related software projects. The staff's evaluation of conformance to IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, based on the applicant's identification of design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, conformance to applicable guidance and IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 20 are adequately addressed.

GDC 21 requires that protection systems be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. The staff evaluated whether GDC 21 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Appendix 7.1-A states that GDC 21 is addressed for protection systems by conformance to IEEE Std 603 criteria, except for Sections 5.4, 6.1, and 7.1. In addition, SRP Section 7.3 identifies that GDC 21 is addressed by conformance to RGs 1.22, 1.47, 1.53, and 1.118 and IEEE Std 379. DCD Tier 2, Revision 9, Table 7.1-1, identifies that the guidelines for periodic testing in RGs 1.22 and 1.118, apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. The bypassed and inoperable status indication conforms to the guidelines of RG 1.47. DCD Tier 2, Revision 9, Section 7.1.2.4, states that the DCIS conforms to the guidelines on the application of the single failure criterion in IEEE Std 379, as supplemented by RG 1.53. DCD Tier 2, Revision 9, Section 7.3, describes the conformance of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems to IEEE Std 603, which is evaluated in Section 7.3.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the design of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems is completed in compliance with IEEE Std 603. In particular, DCD Tier 1, Table 2.2.15-2, provides DAC/ITAAC for verifying that the capability for testing and calibration are met for IEEE Std 603, Sections 5.7 and 6.5. DCD Tier 1, Table 2.2.15-2, also includes DAC/ITAAC for applicable sections of IEEE Std 603, Section 4 for verifying that complete design basis information is identified and implemented for ESF actuation and control systems, VBIF, and ICS DPVIF related software projects. Accordingly, based on the applicant's identification of design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems; conformance to applicable guidance and IEEE Std 603 sections; and their verification in the DAC/ITAAC, the staff finds that the requirements of GDC 21 are adequately addressed.

GDC 22 requires, in the pertinent part, that protection systems be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function. The staff evaluated whether GDC 22 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Appendix 7.1-A states that GDC 22 is addressed for protection systems by conformance to IEEE Std 603, Sections 4, 5.1, 5.3, 5.4, 5.5, 5.6, 6.2, 6.3, 6.8, 7.2, and 8. In addition, SRP Section 7.3 identifies that GDC 22 is addressed by conformance to RG 1.75. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 22 and RG 1.75 apply to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 2, Revision 9, Section 7.3, describes the conformance of ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems to IEEE Std 603, which is evaluated in Section 7.3.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2 includes the DAC/ITAAC for verifying that the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems designs are completed in compliance with IEEE Std 603. In particular, DCD Tier 1, Table 2.2.15-2, provides DAC/ITAAC for IEEE Std 603, Section 5.6. DCD Tier 1, Table 2.2.15-2, also includes DAC/ITAAC for applicable sections of IEEE Std 603, Section 4, to verify that complete design basis information is identified and implemented for ESF actuation and control systems, VBIF, and ICS DPVIF, related software projects. Accordingly, based on the applicant's identification of design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems; conformance to applicable guidance and IEEE Std 603 sections; and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 22 are adequately addressed.

GDC 23 requires that protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if certain conditions are experienced. The staff evaluated whether GDC 23 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Appendix 7.1-A to the SRP states that GDC 23 is addressed for ESF actuation systems, which include the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems, by conformance to IEEE Std 603, Section 5.5. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 23 applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.6, states that the SSLC/ESF fails to a state in which the activated component remains "as-is" to prevent a control-system-induced LOCA. For the same reason, hardware and software failures detected by self-diagnostics do not initiate a signal in a failed SSLC/ESF division. DCD Tier 1, Revision 9, Table 2.2.15-2 and Section 3.2, provide DAC/ITAAC, and Section 3.8 provides ITAAC for verifying that the ESF actuation and control systems, VBIF, and ICS DPVIF design are completed in compliance with IEEE Std 603, Section 5.5. Accordingly, based on the conformance to the applicable guidance and IEEE Std 603 sections and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 23 are adequately addressed.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Appendix 7.1-A states that GDC 24 is addressed for protection systems by conformance to IEEE Std 603, Sections 5.1, 5.6, 5.12, 6.3, 6.6, and 8, particularly Sections 5.6 and 6.3. DCD Tier 2, Revision 9, Section 7.3, describes the conformance of the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems to IEEE Std 603, Sections 5.6 and 6.3, which are evaluated in Section 7.3.3.1 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2 includes the DAC/ITAAC for verifying that the applicable I&C systems' design is completed in compliance with IEEE Std 603, including Sections 5.6 and 6.3. Accordingly, based on the applicant's identification of design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems; conformance to applicable IEEE Std 603 sections; and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 24 are adequately addressed.

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. SRP Appendix 7.1-A states that GDC 29 is addressed by conformance, as applicable, to GDC 20-25 and GDC 28. However, GDC 25, which applies to protection systems requirements for reactivity control malfunctions, and GDC 28, which applies to reactivity control systems, are not applicable to the ESF actuation and control systems evaluated in this section. Accordingly, GDC 29 is addressed by conformance as applicable to GDC 20–24. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the applicable protection and control system designs implement these design criteria. Accordingly, based on the applicant's identification of design bases for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems; conformance to applicable guidance and IEEE Std 603 sections; and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 29 are adequately addressed.

The staff evaluated whether GDC 33, 34, 35, and 38 are adequately addressed. According to SRP Appendix 7.1-A, GDC 33 imposes functional requirements on ESF I&C systems provided to initiate, control, and protect the integrity of reactor coolant makeup systems for protection against small breaks in the RCPB. GDC 33 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 3.1.4.4, identifies that the ICS, ADS, and GDCS provide the reactor coolant makeup functions for the ESBWR to meet the requirements of GDC 33. The ICS is described in DCD Tier 2, Revision 9, Section 5.4.6, and the ADS and GDCS are described in DCD Tier 2, Revision 9, Section 6.3. The staff reviewed the descriptions of their control systems in DCD Tier 2, Revision 9, Section 7.4.4 for the ICS and Section 7.3.1 for the ADS and GDCS and confirmed that these sections identify the corresponding reactor coolant makeup initiation, control, and protection functions. Based on the review of documentation in the above DCD Tier 2, Revision 9, sections, the staff finds that the ESBWR design has provided functions, performance, and reliability necessary to initiate and control the reactor coolant makeup system. Therefore, the staff finds that the safety functions described in GDC 33 are adequately addressed.

According to SRP Appendix 7.1-A, GDC 34 imposes functional requirements on ESF systems provided to initiate, control, and protect the integrity of residual heat removal systems. GDC 34 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 3.1.4.5, identifies that the ICS provides the residual heat removal functions for the ESBWR to meet the requirements of GDC 34. The ICS is described in DCD Tier 2, Revision 9, Section 5.4.6. The staff reviewed the descriptions of its control system in DCD Tier 2, Revision 9, Section 7.4.4, and confirmed that this section identifies the corresponding residual heat removal initiation, control, and protection functions. Based on the review of documentation in the above DCD Tier 2, Revision 9, sections, the staff finds that the ESBWR design has provided functions, performance, and reliability necessary to initiate and control the residual heat removal system. Therefore, the staff finds that the safety functions described in GDC 34 are adequately addressed.

According to SRP Appendix 7.1-A, GDC 35 imposes functional requirements on ESF systems provided to initiate, control, and protect the integrity of the ECCS. GDC 35 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 3.1.4.6, identifies that the ECCS, including the ICS, SLC system, GDCS, and ADS, provides the ECCS functions for the ESBWR to meet the requirements of GDC 35. The ECCS, including the ICS, SLC system, GDCS, and ADS, is described in DCD Tier 2, Revision 9, Section 6.3. The staff reviewed the descriptions of its control system in DCD Tier 2, Revision 9, Section 7.3.1 (for the ADS and GDCS), and confirmed that this section identifies the corresponding ESF-related ECCS initiation, control, and protection functions. Based on the review of documentation in the above DCD Tier 2, Revision 9, sections, the staff finds that the ESBWR design has provided functions, performance, and reliability necessary to initiate and control the ESF systems. Therefore, the staff finds that the safety functions described in GDC 35 are adequately addressed.

According to SRP Appendix 7.1-A, GDC 38 imposes functional requirements on ESF I&C systems provided to initiate, control, and protect the integrity of containment heat removal systems. GDC 38 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 3.1.4.9, identifies that the PCCS provides the containment heat removal functions for the ESBWR to meet the requirements of GDC 38. DCD Tier 2, Revision 9, Section 3.1.4.9, also identifies that while the PCCS has no controls, the ICS provides control functions for the PCCS to fulfill its safety functions. The staff reviewed the descriptions of ICS control system in DCD Tier 2, Revision 9, Section 7.4.4.3 and confirmed that this section has provided necessary containment heat removal, initiation, control, and protection functions to address GDC 38 in the design bases of the ICS. Based on the review of documentation in DCD Tier 2, Revision 9, Section 7.4.4.3, the staff finds that the ESBWR design has provided functions, performance, and reliability necessary to initiate and control the containment heat removal system. Therefore, the staff finds that the safety functions described in GDC 38 are adequately addressed.

In addition, SRP Section 7.3 states that GDC 33, 34, 35, and 38 are addressed by conformance to requirements for testability, operability with onsite and offsite electrical power, and single failures. The single failure and testability requirements correspond to IEEE Std 603, Sections 5.1, 5.7, and 6.5. The staff evaluated conformance of the GDCS and ADS to IEEE Std 603, Sections 5.1, 5.7, and 6.5, in Section 7.3.3.1 of this report and finds it acceptable. The staff evaluated conformance of the ICS to IEEE Std 603, Sections 5.1, 5.7, and 6.5, in Section 7.4.3.1 of this report and finds it acceptable. DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the ICS, ADS, and GDCS designs implement these design bases and conform to IEEE Std 603. For operability with onsite and offsite electrical power, DCD Tier 2, Revision 9, Section 8.1.3, notes that the Q-DCIS, which includes the ICS, ADS, and GDCS, is normally powered by the safety 120-volts ac power distribution system or, if power is lost, by safety batteries for 72 hours. Therefore, these systems are operable using either onsite or offsite power (assuming that only one source is available). Chapter 8 of this report evaluates the safety 120-volts ac power distribution system and batteries. Accordingly, based on the applicant's identification of the necessary reactor coolant makeup, residual heat removal, ECCS, and containment heat removal initiation, control, and protection functions in the design bases of the ICS, ADS, and GDCS, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 33, 34, 35 and 38 are adequately addressed.

As described in section 7.1.1.3.6 of this report, the staff finds that GDCs 41 and 44 are not applicable to the I&C design.

Section 7.1.1.3.7 of this report documents the staff's evaluation of the conformance of the I&C system to the SRM to SECY-93-087. This evaluation includes the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Accordingly, the staff finds that the guidelines of the SRM to SECY-93-087 are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated whether 10 CFR 50.34(f)(2)(v), 10 CFR 50.34(f)(2)(xii), and 10 CFR 50.34(f)(2)(xiv) are adequately addressed. As described in Section 7.1.1.3.4 of this report, the staff evaluated the I&C system design's compliance with 10 CFR 50.34(f)(2)(v), 10 CFR 50.34(f)(2)(xii), and 10 CFR 50.34(f)(2)(xiv). Accordingly, the staff finds that 10 CFR 50.34(f)(2)(v), 10 CFR 50.34(f)(2)(xii), and 10 CFR 50.34(f)(2)(xiv) are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated whether 10 CFR 50.55a(h) is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. The regulations at 10 CFR 50.55a(h) require compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995, which is evaluated in Section 7.3.3.1 and 7.2.3.2 of this report and finds it to be adequately addressed. Accordingly, the staff finds that 10 CFR 50.55a(h) is adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

The staff evaluated whether the applicant met the requirements of 10 CFR 52.47(b)(1). This regulation requires that the application for design certification contain 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. Section 7.2.3 of this report addresses the ITAAC specific to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. The staff's evaluation of conformance to 10 CFR 52.47 in Section 7.1.1.3.4 of this report applies to the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. Therefore, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems.

7.3.4 Conclusion

Based on the above, the staff concludes that the applicant adequately addresses the major design considerations for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems. As discussed in Sections 7.1.1.3.1 through 7.1.1.3.10 of this report and Section 7.3.3 above, the staff concludes for the ESF actuation and control systems, VBIF, ICS DPVIF, and all subsystems that the applicant adequately addresses the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f); 10 CFR 52.47(b)(1); GDC 1, 2, 4, 10, 13, 15, 16, 19 - 24, 29, 33, 34, 35 and 38; and the guidelines of SRM to SECY-93-087. The applicant also identified adequate high-level functions and included sufficient DAC/ITAAC in DCD Tier 1, Revision 9, to verify that the ESF design is completed in compliance with the applicable requirements. The staff also concludes that the requirements of GDC 41 and 44 do not apply to the ESBWR I&C design.

7.4 Safe Shutdown Systems

7.4.1 Regulatory Criteria

The objective of the review of DCD Tier 1, Revision 9, Section 2.2, and DCD Tier 2, Revision 9, Section 7.4, is to confirm that the safe shutdown systems satisfy the requirements of the acceptance criteria and guidelines applicable to safety systems and that they will meet their safety regulatory acceptance criteria, guidelines, and performance requirements. The review of these systems in this section is limited to those features that are unique to safe shutdown and not directly related to accident mitigation. During safe shutdown, reactivity control systems must maintain a subcritical condition of the core, and residual heat removal systems must operate to maintain adequate cooling of the core.

SRP acceptance criteria in SRP Section 7.4, Revision 5,for the safe shutdown systems that are safety systems are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1, 2, 4, 13, 19, 24, 34, 35, and 38; and 10 CFR 52.47(b)(1). The SRP acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152.

SRP acceptance criteria for the safe shutdown systems that are nonsafety systems are based on the relevant requirements of 10 CFR 50.55a(h) and IEEE Std 603, Sections 5.6.3 and 6.3.

7.4.2 Summary of Technical Information

The safe shutdown systems are those I&C systems used to achieve and maintain a safe-shutdown condition of the plant. The safety systems ICS and SLC use natural circulation in the performance of their shutdown functions. I&C design in these systems are part of the Q-DCIS which conforms to the safety criteria (i.e., IEEE Std 603). The two safety RSS panels use manual and the DCIS indication and controls. These two panels are located outside the MCR and are separate from each other. In addition to safety systems, some nonsafety systems are used to perform cold shutdown functions. I&C design in these systems is part of the N-DCIS. For those N-DCIS systems, provision of redundant trains and single failure protection are implemented.

DCD Tier 2, Revision 9, Section 7.4, describes the safe shutdown systems. The safe shutdown systems for the design include the following:

- SLC
- RSS
- RWCU/SDC
- ICS
- HP CRD IBF

7.4.2.1 *Standby Liquid Control System*

The SLC system, an ECCS, provides (1) a diverse backup means to shut down the reactor from full power to subcritical condition, using soluble boron injection, and maintain the reactor subcritical while the reactor is brought to a cold shutdown condition, and (2) system actuation upon receipt of manual and automatic initiation signals in response to either ATWS events or DBEs requiring ECCS operation.

The SLC system contains two identical and separate trains. Each train provides 50-percent injection capacity. The SLC also includes a nonsafety nitrogen charging subsystem that includes a liquid nitrogen tank, vaporizer, and high pressure pump for initial accumulator charging and makeup for the normal system losses during normal plant operation.

Four divisions of safety sense and command logic are used for automatic SLC initiation and for automatic SLC accumulator isolation. Redundant SLC accumulator level and pressure instrumentation is provided to monitor system performance. Valve position indication and continuity monitoring of the SLC injection squib valves are provided in the MCR. Safety SLC components are designed for the environmental conditions applicable to their location. Safety SLC components are also designed to preclude adverse interaction from the nonsafety portion of the system.

The SLC is initiated automatically as part of the ECCS to provide mitigation for LOCA events. The SLC receives an actuation command following a confirmed LOCA signal plus a 50-second time delay after a sustained RPV Level 1 signal for 10 seconds. The SLC also receives a diverse ECCS initiation signal from the DPS.

The SLC also starts automatically upon an ATWS mitigation signal persisting for 180 seconds. The ATWS mitigation logic performs the diverse emergency shutdown function. The ATWS/SLC logic uses sensors, hardware, and software platforms diverse from the SSLC/ESF, the RPS, and the DPS hardware/software platforms.

To avoid boron dilution during SLC operation, the SLC system logic transmits an isolation signal to the RWCU/SDC via the LD&IS. To avoid the injection of nitrogen into the RPV system, four divisional, safety level sensors per SLC accumulator are used to provide automatic isolation of series accumulator shutoff valves upon a voted 2/4 low accumulator level. The SLC system processors of the ATWS/SLC independent control logic platform perform the shutoff valve isolation logic. Accumulator temperature, solution level, and accumulator pressure are indicated locally inside the accumulator room. Boron injection and shutoff valve position status is provided in the MCR.

7.4.2.2 *Remote Shutdown System*

The safety RSS provides operators with the means to safely shut down the reactor from a place outside the MCR if it becomes uninhabitable. The RSS provides remote control of the systems needed to bring the reactor to a hot shutdown after a scram. The RSS also provides the subsequent capability to bring the plant to and maintain a cold shutdown condition.

The RSS has two redundant and independent panels. All parameters displayed and/or controlled from Division 1 and Division 2 in the MCR also are displayed and/or can be controlled from any of the two RSS panels. Each panel contains the following:

- Division 1 manual scram switch
- Division 2 manual scram switch
- Division 1 manual MSIV isolation switch
- Division 2 manual MSIV isolation switch
- Division 1 safety VDUs
- Division 2 safety VDUs
- PIP A nonsafety VDUs
- PIP B nonsafety VDUs
- Nonsafety communications equipment

All data from the Q-DCIS and the N-DCIS networks are available for display on the RSS panels. Because the VDUs on the RSS panels are connected to the Q-DCIS or the N-DCIS through the same networks serving corresponding VDUs at the MCR, all Division 1 and 2 safety and nonsafety display/control functions at the MCR also are available at the RSS panels.

The two RSS panels are located in different rooms inside the reactor building. Each RSS panel room has a sliding fire door with a minimum fire rating of 3 hours. The RSS panel room environment is typically similar to the MCR environment. Access to and use of the RSS panels are administratively controlled.

The RSS provides sufficient redundancy in its control and monitoring capability to accommodate a single failure in the interfacing systems, a single failure in the RSS controls, and the event that caused the MCR evacuation. The RSS is designed such that any failure within it does not degrade the capability of interfacing safety systems.

7.4.2.3 *Reactor Water Cleanup/Shutdown Cooling System*

The RWCU/SDC is a nonsafety system that provides cooling for the reactor to reach cold shutdown condition. There are two redundant RWCU/SDC trains. The loss of one complete RWCU/SDC train could extend the time needed for the reactor to reach cold shutdown condition. The RWCU/SDC system is one of the dual-redundant PIP systems whose instrumentation belongs to the N-DCIS.

The RWCU/SDC system performs three basic plant functions—(1) it provides a continuous purifying treatment of the reactor coolant during startup, normal operation, cooldown, hot standby, and shutdown modes of plant operation; (2) it removes core decay heat in conjunction with the main condenser or the isolation condensers during plant shutdown modes; and (3) along with the feedwater system, it provides reactor coolant heatup during cold startup. The I&C portion of the RWCU/SDC system maintains the process conditions within the limits necessary to control the system and satisfy its design bases.

7.4.2.4 *Isolation Condenser System*

The ICS removes core decay heat from the reactor following any of the following events:

- SBO
- ATWS event
- LOCA

The ICS is capable of passive decay heat removal and achieving and maintaining safe, stable conditions for at least 72 hours without operator action following non-LOCA events. Operator action is credited after 72 hours to refill isolation condenser pools or initiate nonsafety shutdown cooling.

The ICS is one of the ESF systems whose instrumentation belongs to the Q-DCIS. The ICS consists of four independent trains, each containing an isolation condenser that condenses steam on the tube side and transfers heat to the IC/PCCS pool, which is vented to the atmosphere. The isolation condenser, connected by piping to the RPV, is placed at an elevation above the source of steam (vessel) and, when the steam is condensed, the condensate is returned to the vessel via a condensate return pipe. The steam side connection between the vessel and the isolation condenser is normally open, and the condensate line is normally closed. This allows the isolation condenser and drain piping to fill with condensate, which is maintained at a subcooled temperature by the pool water during normal reactor operation. The isolation condenser is started into operation by opening condensate return valves and draining the condensate to the reactor, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler pool water.

The ICS is designed to operate from safety power sources. The system instrumentation is powered by four divisionally separated sources of safety power. The ICS uses 2/4 logic for automatic operation or isolation of each of the four separate isolation condenser trains. The actuating logic and actuator power for the inner isolation valves for the four ICS trains are on two safety 120 volt ac divisional power sources which are different than the two divisional power sources for the outer isolation valves.

Each of the four ICS trains has three of the four safety power sources. Consequently, the loss of 2/4 safety power supplies does not result in the loss of any one ICS train. However, second and third sources of safety power are provided to operate the ICS automatic venting system during long-term ICS operation; otherwise, the manually controlled backup venting system, which uses one of the divisional power sources starting the ICS, can be used for long-term operation. If the three safety power supplies used to start an individual ICS train fail, then the isolation condenser would automatically start because of the “fail open” actuation of the condensate return bypass valves upon loss of electrical power to the solenoids controlling its nitrogen-actuated valves.

The ICS receives an actuation command following a confirmed LOCA signal after a time delay corresponding to the first depressurization valve actuation.

The ICS starts operating automatically upon high reactor pressure, low reactor water level (Level 2) with time delay, low reactor water level (Level 1), loss of power generation buses, loss of feedwater flow in reactor run mode, or MSIV position indication (indicating closure) whenever the reactor mode switch is in the run position. Each ICS train also can be manually initiated, enabling the operator to stop any individual ICS train whenever the RPV pressure is below a reset value override to the ICS automatic actuation signal following MSIV closure.

The residual heat removal function of the safety ICS is further backed up by the safety ESF combination of ADS, PCCS, and GDCS; by the nonsafety RWCU/SDC loops; or by the makeup function of the CRD system operating in conjunction with SRVs and the suppression pool cooling system (SPCS). The DPS provides diverse nonsafety signals for ICS actuation and other ICS functions.

7.4.2.5 *High Pressure Control Rod Drive Isolation Bypass Function*

The CRD hydraulic system supplies high pressure makeup water to the reactor vessel in response to a low RPV water level (Level 2) condition or in the event the GDCS fails to inject following a LOCA. The HP CRD IBF is designed to mitigate the beyond design basis failure of the GDCS to inject following a LOCA. The HP CRD IBF has the following design bases:

- Using safety logic inputs, the normally closed HP CRD isolation bypass valves are opened automatically upon failure of the GDCS to inject water in to the reactor.

- Nonsafety manual control of the HP CRD isolation bypass valve is provided and isolation bypass valve positions are displayed in the MCR.

- Divisional instrumentation performing the HP CRD IBF logic is powered by the associated safety divisional power supply.

- Bypass of a division of sensors is annunciated in the MCR.

- The HP CRD IBF logic executed in the ICP is independent and diverse from the SSLC/ESF.

7.4.3 Staff Evaluation

The staff reviewed the safe shutdown systems in accordance with SRP Section 7.4. The staff also used acceptance criteria in SRP Section 7.1, SRP Table 7-1, SRP Appendix 7.1-A, and SRP Appendix 7.1-C, as directed by SRP Section 7.4. Section 7.4.1 of this report describes the acceptance criteria used as the basis for the staff's review of the safe shutdown systems.

SRP Section 7.1 describes the procedures to be followed in reviewing any I&C system. SRP Section 7.4 highlights specific topics that should be emphasized in reviewing the safe shutdown systems, and Section 7.4.3.1 of this report addresses the specific topics. The staff included the review of the DAC/ITAAC during the review of this section because of their significant role in determining the conformance of safe shutdown systems to all requirements. GDC 44 requires a system to transfer heat from SSCs important to safety to an ultimate heat sink. According to SRP Appendix 7.1-A, GDC 44 imposes functional requirements on the safe shutdown systems.

As described in Section 7.1.1.3.1 of this report, the DCD does not provide the required safe shutdown system design information to comply with IEEE Std 603. Instead, the applicant has included the DAC/ITAAC in DCD Tier 1, Revision 9, Section 2.2.15, to confirm that the completed safe shutdown systems' design complies with IEEE Std 603. DCD Tier 1, Revision 9, Section 2.2.15, also includes an ITAAC applicability table (Table 2.2.15-1), which identifies the applicability of the IEEE Std 603 criteria DAC/ITAAC to the safe shutdown systems. The staff has accepted the DAC approach to addressing compliance with IEEE Std 603. The staff's evaluation of conformance to IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the safe shutdown systems.

In RAI 7.1-99, the staff asked the applicant to clarify the applicability of IEEE Std 603 criteria in a consistent manner throughout DCD Tier 2, Chapter 7. The applicant submitted a response to RAI 7.1-99, along with responses to RAIs 7.1-100, 7.1-101, and RAI 14.3-265 S01, all of which are incorporated in DCD Revision 8. Section 7.1.1.3.1 of this report provides additional discussion of the resolution of these RAIs. With regard to the staff's evaluation of the safe shutdown systems, the applicant significantly revised DCD Tier 2, Tables 7.1-1 and 7.1-2, to clearly identify the applicability of the IEEE Std 603 criteria to each safe shutdown system. Concurrently, the applicant revised or removed many references to the IEEE Std 603 criteria from the discussions of safe shutdown systems in DCD Tier 2, Section 7.4 to address the consistency concerns. Accordingly, the staff updated or removed statements regarding the applicability of IEEE Std 603 criteria provided in the SER with open items from the remainder of Section 7.4.3 of this report to be consistent with DCD Revision 8.

7.4.3.1 *Evaluation of Safe-Shutdown Systems Conformance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.4, the following major design considerations should be emphasized in the safe shutdown systems review:

(1) Independence (IEEE Std 603, Sections 5.6 and 6.3)

The staff evaluated whether Sections 5.6 and 6.3 of IEEE Std 603 are adequately addressed for the safe shutdown systems. Section 7.1.1.3.10 of this report evaluates conformance with these criteria. The staff finds that Sections 5.6 and 6.3 are adequately addressed based on their inclusion in the safety systems design basis and their verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the safe shutdown systems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that Sections 5.6 and 6.3 of IEEE Std 603 apply to the SLC, ICS, and HP CRD IBF that make up the applicable safe shutdown systems. Accordingly, based on the inclusion of IEEE Std 603, Sections 5.6 and 6.3, in the applicable safe shutdown systems' design basis and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed for the safe shutdown systems.

(2) Use of Digital Systems (IEEE Std 7-4.3.2)

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed for the safe shutdown systems. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2 concerning the use of digital systems. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in Appendix 7.1-D to the SRP. The staff's evaluation of conformance to IEEE Std 7-4.3.2 in Section 7.1.1.3.10 of this report applies to the safe shutdown systems.

NEDE-33226P describes the software development activities and NEDE-33245P describes the software QA activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff's evaluation of the software development and software QA activities. Accordingly, the staff finds that use of digital systems is adequately addressed for the applicable safe shutdown systems.

(3) Periodic Testing (IEEE Std 603, Sections 5.7 and 6.5)

The staff evaluated whether conformance with IEEE Std 603, Sections 5.7 and 6.5, is adequately addressed for the safe shutdown systems. Section 7.1.1.3.10 of this report evaluates conformance with these criteria. The staff finds that Sections 5.7 and 6.5 are adequately addressed based on their inclusion in the safety systems' design basis and their verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. DCD Tier 1, Revision 9,

Table 2.2.15-1, identifies that Sections 5.7 and 6.5 of IEEE Std 603 apply to the SLC, ICS, and HP CRD IBF that make up the relevant safe shutdown systems. Accordingly, based on the inclusion of IEEE Std 603, Sections 5.7 and 6.5, in the applicable safe shutdown systems' design basis and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that IEEE Std 603, Sections 5.7 and 6.5, are adequately addressed for the safe shutdown systems.

(4) Remote Shutdown Capability

The accident analysis in DCD Tier 2, Revision 9, Chapter 15, does not analyze performing a reactor shutdown remote from the MCR. Specific scenarios have not been identified for which the adequacy of shutdown capability from the RSS is evaluated. However, smoke resulting from a fire in the MCR has long been recognized as the event that could force the evacuation of the MCR and result in a need to shut down the reactor remotely from the MCR. RG 1.189 establishes the bases for safe shutdown with respect to fire protection. On the basis of DCD Tier 2, Revision 9, Sections 15.5.6.2 and 15.5.6.3, which provide the assumptions and results of the safe shutdown fire analysis, only a manual scram of the plant from the MCR is required to reach and maintain Mode 3 (hot shutdown).

The staff evaluated whether the design provides for control in locations remote from the MCR that may be used for manual control and alignment of the safe shutdown system equipment needed to achieve and maintain hot and cold shutdown. The staff also evaluated whether this control equipment is capable of operating independently of (i.e., without interaction with) the equipment in the MCR.

DCD Tier 2, Revision 9, Section 7.4, states that the RSS is a safety system used to provide operators with the means to safely shut down the reactor from a place outside the MCR, if the MCR becomes uninhabitable. The RSS provides remote control of the systems that are needed to bring the reactor to a hot shutdown condition after a scram. The RSS also provides the subsequent capability to bring the plant to and maintain the reactor plant in a cold shutdown condition. The staff finds this acceptable.

The staff evaluated whether the design of the RSS provides appropriate displays to enable the operator to monitor the status of the shutdown. SRP Section 7.4 states that typical RSS displays include reactor vessel water level and pressure, suppression pool level and temperature, isolation condenser level indication for tanks involved in shutdown, and shutdown system diagnostic instrumentation.

DCD Tier 2, Revision 9, Section 7.4, states that the RSS has two redundant and independent panels. All parameters that are displayed and/or controlled from Division 1 and Division 2 in the MCR are also displayed and/or can be controlled from any of the two RSS panels.

The staff evaluated whether the remote shutdown capability is able to accommodate expected plant response following a reactor trip, including protective system actions that could occur as a result of plant cooldown. DCD Tier 2, Revision 9, Section 7.4.2.3, states the following:

> The RSS provides instrumentation and controls (I&C) outside the MCR to allow prompt hot shutdown of the reactor after a scram and to maintain safe conditions during hot shutdown. It also provides capability for subsequent cold shutdown of the reactor through the use of suitable operating procedures.

The staff finds this acceptable.

The staff evaluated whether access to the RSS is under administrative controls. DCD Tier 2, Revision 9, Section 7.4.2.2.1, states, “Access to and use of the RSS panels is administratively controlled.” The staff finds this acceptable.

The staff evaluated whether the equipment in the RSS is designed to the same standards as the corresponding equipment in the MCR. DCD Tier 2, Revision 9, Section 7.4.2.3, states, “The RSS is classified as a safety system that can control safety systems or equipment,” and Section 7.4.2.2.1 states, “All parameters displayed and/or controlled from Division 1 and Division 2 in the MCR also are displayed and/or can be controlled from any of the two RSS panels.” The staff finds this acceptable.

The staff evaluated whether the RSS-control transfer devices should be located remote from the MCR and whether their use should initiate an alarm in the control room. DCD Tier 2, Revision 9, Section 7.4.2.2.3, states the following:

> When evacuation of the MCR is necessary, the reactor is manually scrammed. If there has been no loss of off-site power, the turbine bypass valves automatically control reactor pressure, and the reactor feedwater system automatically maintains RPV water level. These functions will remain operable because the safety and non-safety controllers are not located in the same fire area as the MCR nor are they affected by the adverse impacts on the MCR VDUs and switches after an MCR evacuation; as a result, reactor cooldown is achieved through the normal heat sinks. However, if the reactor feedwater system is not available due to loss of off-site power, control of the CRD system from the RSS may be utilized. Control of the high pressure makeup injection capability of the CRD system ensures that the RPV water level remains above the ADS trip setpoint and above the elevation of the RWCU/SDC mid-vessel suction line nozzle. The ICS automatically controls reactor pressure. ICS operation is not affected by an MCR evacuation. With the ICS in operation, the isolation condensers provide initial decay heat removal, and further reactor cooldown is achieved from the RSS panels using the RWCU/SDC.

Therefore, no remote transfer devices are necessary for the design. The staff finds this acceptable.

The staff evaluated whether the location is consistent with the procedures for remote, alternative, and dedicated shutdown, as appropriate. DCD Tier 2, Revision 9, Section 7.4.2.2.1, states, “The two RSS panels are located in different rooms inside the Reactor Building (RB). Each RSS Panel room has a sliding fire door with a minimum fire rating of 3 hours. The RSS panel room environment typically is similar to the MCR environment.” The staff finds this acceptable.

The staff evaluated whether, in cases in which the control functions are transferred between the control room and the RSS, the design maintains parameter indications such that the operators at the control room and the RSS both have access to the same parameters that are being relied upon. DCD Tier 2, Revision 9, Section 7.4.2.2.1, states, “All parameters displayed and/or controlled from Division 1 and Division 2 in the MCR also are displayed and/or can be controlled from any of the two RSS panels.” Therefore, transfer of control functions is not necessary for the design. The staff finds this acceptable.

If the MCR evacuation is necessary, the remote shutdown panels provide complete redundancy in terms of control and monitoring for safe-shutdown functions. The transfer of operation from the MCR to the remote shutdown panel is not required since the remote shutdown panels are designed to have all the functions available at the MCR. The MCR is located in the control building, and remote shutdown panels are located in separate fire areas in the reactor building. The MCR has its own dedicated ventilation system, and the remote shutdown panel area ventilation system will use the reactor building ventilation system. The safety and nonsafety electrical cabinets are located in the separate DCIS rooms, which are in different fire areas. Communications between the MCR, remote shutdown panels, and the DCIS rooms use fiber optic cables. The HFE process ultimately decides the hard-wired controls in the MCR or in the RSS.

In DCD Tier 1, Revision 9, Section 2.2.6, the applicant documented the RSS design requirements and the ITAAC for the RSS. The RSS is a safety, seismic Category I system. The RSS has two redundant, independent panels, and panels are located in two separate rooms in different divisional quadrants of the reactor building. Safety systems in each RSS panel receive power from divisionally separate safety power supplies; nonsafety systems in each RSS panel receive power from nonsafety power supplies. Based on the design described in DCD Tier 2, Revision 9, Section 7.4.2, and DCD Tier 1, Revision 9, Section 2.2.6, the staff finds the I&C design for the RSS acceptable.

(5) Safe Shutdown

The staff evaluated whether the single failure criterion, IEEE Std 603, Section 5.1, is adequately addressed for the safe shutdown systems. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, Section 5.1. The staff finds that Section 5.1 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that Section 5.1 of IEEE Std 603 applies to the SLC, ICS, and HP CRD IBF that make up the applicable safe shutdown systems. Accordingly, based on the inclusion of IEEE Std 603, Section 5.1, in the applicable safe shutdown systems design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that IEEE Std 603, Section 5.1, is adequately addressed for the safe shutdown systems.

The staff evaluated whether the safe shutdown systems provide the required capacity and reliability to perform intended safety functions on demand in conformance with IEEE Std 603, Section 5. The staff previously evaluated conformance with IEEE Std 603, Section 5, in Section 7.1.1.3.10 of this report and finds it to be acceptable. The staff's evaluation of conformance to IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the safe shutdown systems. Accordingly, the staff finds that IEEE Std 603, Section 5, is adequately addressed for the safe shutdown systems.

The staff evaluated whether the safe shutdown systems provide the required capacity to function during and after DBEs, such as earthquakes and AOOs, in conformance with IEEE Std 603, Sections 5.4 and 5.5. The staff evaluated conformance with IEEE Std 603, Section 5.4 for the safe shutdown systems that are safety systems. Section 7.1.1.3.10 of this report evaluates conformance with this criterion, and the staff finds that Section 5.4 is adequately addressed based on its inclusion in the safety systems' design basis and verification of the EQ in the DCD Tier 1, Revision 9, Section 3.8, ITAAC. This evaluation applies to the safe shutdown systems that are safety systems. Accordingly, the staff finds that Section 5.4 is adequately addressed for the safe shutdown systems that are safety systems.

The staff evaluated conformance with IEEE Std 603, Section 5.5, for the safe shutdown systems that are safety systems. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.5 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 2.2.15 and Section 3.2, DAC/ITAAC. This evaluation applies to the safe shutdown systems that are safety systems. Accordingly, the staff finds that IEEE Std 603, Section 5.5, is adequately addressed for the safe shutdown systems that are safety systems.

The staff evaluated whether the safe shutdown systems operate with onsite electric power available (assuming that offsite power is not available) and with offsite electric power available (assuming that onsite power is not available). Chapter 8 of this report evaluates electric power systems. Additionally, DCD Tier 2, Revision 9, Sections 7.4.1.2.1, 7.4.2.2.2, 7.4.3.2.2, and 7.4.4.3 state the following:

- "Power for the safety functions of the SLC system is derived from safety 120 VAC electrical systems UPS. Divisional assignments are made to ensure the availability of each SLC system loop, assuming one safety division of power is not in service in addition to a single active failure. Additionally, a squib initiator in each loop is activated by the DPS as part of the D3 strategy. To avoid adverse interaction, electrical isolation is maintained between the safety divisions, and between the safety divisions and the DPS."

- "The RSS panel is powered from buses supplied by uninterruptible safety and non-safety 120 VAC systems."

- "The RWCU/SDC pumps are supplied from separate and preferred power sources. The power supplies are automatically switched to dual on-site standby diesel-generators following the loss of preferred power (LOPP)."

- "The actuating logic and actuator power for the inner isolation valves for the four ICS trains are on two safety 120 VAC divisional power sources UPS different from the two divisional power sources for the outer isolation valves."

In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for IEEE Std 603, Section 8.1, are required for the safe shutdown systems. The staff finds that the electric power supply is adequately addressed for the safe shutdown systems.

The staff evaluated whether the safe shutdown systems provide the capability to be tested during reactor operation in conformance with IEEE Std 603, Sections 5.7 and 6.5. As described in Item (3) above, based on the inclusion of IEEE Std 603, Sections 5.7 and 6.5, in the safe shutdown systems design basis and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that IEEE Std 603, Sections 5.7 and 6.5, are adequately addressed for the safe shutdown systems.

7.4.3.2 *Evaluation of Safe Shutdown System Conformance with Acceptance Criteria - IEEE Std 603 and IEEE Std 7-4.3.2 Criteria*

SRP Section 7.4 states that the safe shutdown systems' design should be evaluated for conformance to IEEE Std 603. This section evaluates conformance with IEEE Std 603 criteria not previously evaluated in Section 7.4.3.1 of this report. The applicable safe shutdown systems with regard to IEEE Std 603 are the SLC, ICS, and HP CRD IBF.

The staff evaluated the safe shutdown systems design basis to determine whether IEEE Std 603, Section 4, is adequately addressed using SRP Appendix 7.1-C, Section 4. For completeness, the SRP states, "As a minimum each of the safety system design basis aspects identified in IEEE Std 603, Sections 4.1 through 4.12 should be addressed." Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 4 is adequately addressed based on its inclusion in the safety system design basis and the verification of applicable criteria in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the safe shutdown systems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for the relevant criteria of Section 4 apply to the safe shutdown systems.

DCD Tier 2, Revision 9, Section 7.4 identifies individual parameters that determine operation of the safe shutdown systems. As mentioned previously, NEDE-33226P and NEDE-33245P, as part of the software life cycle process, define a process by which plant performance requirements, including response times, under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. Accordingly, the staff finds that Section 4 is adequately addressed for the safe shutdown systems.

The staff evaluated conformance with IEEE Std 603, Sections 5.2, 5.9, 5.10, 5.11, 5.12, 6.1, 6.2. 6.4, 6.6, 6.7, 6.8, 7.1, 7.2, 7.3, 7.4, 7.5, 8.1, 8.2, and 8.3. Section 7.1.1.3.10 of this report evaluates conformance with these criteria. The staff finds that these criteria are adequately addressed based on their inclusion in the safety system design basis and their verification in the DCD Tier 1, Revision 9, Section 2.2.15, DAC/ITAAC. This evaluation applies to the safe shutdown systems. In addition, DCD Tier 1, Revision 9, Table 2.2.15-1, identifies that the DAC/ITAAC for these criteria apply to the safe shutdown systems. Accordingly, the staff finds that conformance to IEEE Std 603, Sections 5.2, 5.9, 5.10, 5.11, 5.12, 6.1, 6.2. 6.4, 6.6, 6.7, 6.8, 7.1, 7.2, 7.3, 7.4, 7.5, 8.1, 8.2, and 8.3 are adequately addressed for the safe shutdown systems.

The staff evaluated whether the quality criterion, IEEE Std 603, Section 5.3, is adequately addressed for the safe shutdown systems that are safety systems. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.3 is adequately addressed based on the inclusion of IEEE Std 7-4.3.2, Section 5.3, in the safety systems' design basis and the verification of the software development activities in the DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC. In addition, the applicant stated that the quality assurance program conforms to GDC 1. Chapter 17 of this report evaluates adequacy of the quality assurance program. These evaluations apply to the safe shutdown systems that are safety systems. DCD Tier 2, Revision 9, Section 7.1.6.6.1.4, also discusses the applicability of this criterion to the Q-DCIS design. Accordingly, based on the applicant's use of an acceptable software development process, as evaluated in Sections 7.1.2.3 and 7.1.1.3.10.2 of this report, and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that Section 5.3 of IEEE Std 7-4.3.2 and Section 5.3 of IEEE Std 603 are adequately addressed for the safe shutdown systems that are safety systems.

The staff evaluated whether IEEE Std 603, Section 5.8, is adequately addressed. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that the criterion is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 3.3, DAC/ITAAC. This evaluation applies to the safe shutdown systems that are safety systems. In addition, Section 5.8, "Information Displays," is part of system testing and inoperable surveillance. DCD Tier 2, Revision 9, Chapter 18, describes the HFE design process to design information displays, which is evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification applies to the safe shutdown systems that are safety systems and includes verifying the inventory of displays for manually controlled actions, system status indications, and indications of bypasses. Accordingly, the staff finds that IEEE Std 603, Section 5.8, is adequately addressed for the safe shutdown systems that are safety systems.

The staff evaluated conformance with IEEE Std 603, Section 5.13. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The multi-unit station criteria do not apply to the standard single unit plant design submitted for NRC certification, as stated in DCD Tier 2, Revision 9, Section 7.1.6.6.1.14. The staff determines that IEEE Std 603, Section 5.13, is not applicable to design certification.

The staff evaluated conformance with IEEE Std 603, Section 5.14. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.14 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, Section 3.3, DAC/ITAAC. This evaluation applies to the safe shutdown systems that are safety systems. Accordingly, the staff finds that IEEE Std 603, Section 5.14, is adequately addressed for the safe shutdown systems that are safety systems.

The staff evaluated conformance with IEEE Std 603, Section 5.15. Section 7.1.1.3.10 of this report evaluates conformance with this criterion. The staff finds that Section 5.15 is adequately addressed based on its inclusion in the safety systems' design basis and its verification in the DCD Tier 1, Revision 9, DAC/ITAAC for IEEE Std 603, Sections 5.1; DCD Tier 1, Revision 9, Section 3.2, DAC/ITAAC; and DCD Tier 1, Revision 9, Section 3.6, ITAAC. This evaluation applies to the safe shutdown systems that are safety systems. Accordingly, the staff finds that IEEE Std 603, Section 5.15, is adequately addressed for the safe shutdown systems that are safety systems.

7.4.3.3 *Evaluation of Safe-Shutdown System Compliance with GDC*

The staff reviewed the acceptance criteria for safe shutdown systems in accordance with SRP Section 7.4 and SRP Appendix 7.1-A. For several of the GDC, compliance can be satisfied by meeting IEEE Std 603 requirements, which the staff evaluated in the previous two sections. Compliance with IEEE Std 603 is briefly discussed along with the relevant GDC, including the use of DAC, consistent with both SRP sections.

GDC 1 requires quality standards and maintenance of appropriate records. 10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The staff evaluated whether GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed for the safe shutdown systems in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 1 and 10 CFR 50.55a(a)(1) apply to the safe shutdown systems. The staff evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1) and GDC 1 in Sections 7.1.1.3.3 and 7.1.1.3.6 of this report applies to the safe shutdown systems. Accordingly, based on the review of

Revision 9, DCD information, and the applicant's identification of design bases for the safe shutdown systems and their verification in the DCD Tier 1, Revision 9, ITAAC, the staff finds that the requirements of GDC 1 are adequately addressed for the safe shutdown systems.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the safe shutdown systems. SRP Section 7.4 identifies that GDC 2 and 4 are addressed by the identification of those systems and components for the safe shutdown systems designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles in the design bases. SRP Section 7.4 also identifies that GDC 2 and 4 are addressed by the review of the qualification program in DCD Tier 2, Sections 3.10 and 3.11. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 2 and 4 apply to the safe shutdown systems. DCD Tier 2, Revision 9, Table 3.2-1, identifies that the safe shutdown systems that are safety systems are designed as seismic Category I systems. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Chapter 3 of this report. In DCD Tier 1, Revision 9, Table 3.8-1, Items 1 and 3, include the ITAAC for the applicant to verify the EQ of safety electrical and digital I&C equipment. The evaluation of GDC 2 and GDC 4 in Section 7.1.1.3.6 of this report further addresses these topics and applies to the safe shutdown systems. Accordingly, based on the applicant's identification of EQ programs consistent with the design bases for the safe shutdown systems and their verification in the DCD Tier 1, Revision 9, ITAAC, the staff finds that the requirements of GDC 2 and 4 are adequately addressed.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 13 and 19 are adequately addressed for the safe shutdown systems. Section 7.1.1.3.6 of this report provides the evaluation of conformance with GDC 19 with the exception of the safe shutdown systems' support functions necessary for shutting down the reactor and remote shutdown capability. SRP Section 7.4 identifies that GDC 13 and 19 are addressed by the review of I&C required for safe shutdown and the review of I&C within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including a shutdown following an accident. GDC 19 also requires that equipment at appropriate locations outside the control room be provided (1) with a design capability for prompt, hot shutdown of the reactor, including necessary I&C to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

The RWCU/SDC is the nonsafety shutdown system that has the capability of bringing the reactor to safe shutdown and cold shutdown during normal operations. The SLC and ICS are safety systems that provide for safe shutdown during anticipated occurrences and accidents. DCD Tier 2, Revision 9, Sections 7.4.3.2.2, 7.4.1.2, 7.4.4.3, and 7.4.5.2, specify the automatic and manual initiation controls of the RWCU/SDC, SLC, ICS, and HP CRD IBF, respectively. DCD Tier 2, Revision 9, Sections 7.4.3.5, 7.4.1.5, 7.4.4.5, and 7.4.5.5, specify the status indication and the alarms provided for the RWCU/SDC, SLC, ICS, and HP CRD IBF, respectively. In combination with the following identified interrelated processes to complete the design of the monitoring capability and control room controls for the safe shutdown systems, the staff finds that the safe shutdown systems provide the I&C needed to maintain variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems within prescribed operating ranges during plant shutdown. In addition, the safe shutdown systems provide within the MCR the I&C needed to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including a shutdown following an accident. NEDE-33226P and NEDE-33245P, as part of a software life cycle process, define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing an HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. These verifications apply to the safe shutdown systems and include verification of the controls for manual initiation and control of safe shutdown functions necessary to support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

DCD Tier 2, Revision 9, Section 7.4.2.5, specifies that the parameters displayed and/or controlled from Division 1 and Division 2 in the MCR also are displayed or can be controlled from either of the RSS panels. The staff's conclusions regarding the MCRs monitoring and controls apply to the remote shutdown capability. Section 7.4.3.1, Item (4), of this report evaluates the remote shutdown capability, which the staff finds to be acceptable. Therefore the staff finds that equipment provided at appropriate locations outside the MCR includes (1) a design capability for prompt, hot shutdown of the reactor, including necessary I&C to maintain the unit in a safe condition during hot shutdown, and (2) a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. Accordingly, based on the identified monitoring capabilities and controls, the defined processes for completing their design, and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for the safe shutdown systems. Appendix 7.1-A to the SRP states that GDC 24 is addressed for safety systems by conformance to IEEE Std 603, Sections 5.1, 5.6, 5.12, 6.3, 6.6, and 8, particularly Sections 5.6 and 6.3. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 24 applies to the safe shutdown systems. DCD Tier 2, Revision 9, Section 7.4, describes the conformance of safe shutdown systems to IEEE Std 603, Sections 5.6 and 6.3, which are evaluated in Section 7.1.1.3.10 of this report. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the applicable I&C systems' design is completed in compliance with IEEE Std 603, including Sections 5.6 and 6.3. Accordingly, based on the applicant's identification of design bases for the safe shutdown systems, conformance to applicable IEEE Std 603 sections, and their verification in the DCD Tier 1, Revision 9, ITAAC, the staff finds that the requirements of GDC 24 are adequately addressed.

The staff evaluated whether GDCs 34, 35, and 38 are adequately addressed. According to SRP Appendix 7.1-A, GDC 34 imposes functional requirements on safe shutdown systems provided to initiate, control, and protect the integrity of residual heat removal systems. GDC 34 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 3.1.4.5, states that GDC 34 applies to the ICS, which is described in DCD Tier 2, Revision 9, Section 5.4.6. DCD Tier 2, Revision 9, Section 7.4.4, identifies the corresponding residual heat removal initiation, control, and protection functions in the design bases. According to SRP Appendix 7.1-A, GDC 35 imposes functional requirements on safe shutdown systems provided to initiate, control, and protect the integrity of the ECCS. GDC 35 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 3.1.4.6, states that GDC 35 applies to the ECCS, including the ICS, SLC system, GDCS, and ADS, as described in DCD Tier 2, Revision 9, Section 6.3. DCD Tier 2, Revision 9, Sections 7.4.4 (ICS) and 7.4.1 (SLC system), identify the corresponding safe-shutdown-related ECCS initiation, control, and protection functions in the design bases. According to SRP Appendix 7.1-A, GDC 38 imposes functional requirements on safe shutdown systems provided to initiate, control, and protect the integrity of containment heat removal systems. GDC 38 also requires that necessary I&C systems are operable using either onsite or offsite power (assuming that only one source is available). DCD Tier 2, Revision 9, Section 7.4.4.3, identifies the ICS containment heat removal initiation, control, and protection functions in the design bases.

In addition, SRP Section 7.4 identifies that GDC 34, 35, and 38 are addressed by review for conformance to requirements for testability, operability with onsite and offsite electrical power, and single failures. The single failure and testability requirements correspond to IEEE Std 603, Sections 5.1, 5.7, and 6.5. The staff evaluated conformance of the ICS and SLC system to IEEE Std 603, Sections 5.1, 5.7, and 6.5, in Section 7.4.3.1 of this report and finds that they are adequately addressed. In DCD Tier 1, Revision 9, Sections 2.2.15, 3.2, 3.3, and 3.8, include the DAC/ITAAC for the applicant to verify that the ICS and the SLC system design implements these design bases and conforms to IEEE Std 603.

For operability with onsite and offsite electrical power, DCD Tier 2, Revision 9, Section 8.1.3, identifies that the Q-DCIS, which includes the ICS and SLC system I&C systems, is powered by the safety power distribution system normally or by safety batteries for 72 hours if power is lost. Therefore, these systems are operable using either onsite or offsite power (assuming that only one source is available). Chapter 8 of this report evaluates the safety power distribution system and batteries. Accordingly, based on the applicant's identification of necessary residual heat removal, ECCS, and containment heat removal initiation, control, and protection functions in the design bases of the ICS and their verification in the DCD Tier 1, Revision 9, DAC/ITAAC, the staff finds that the requirements of GDC 34, 35, and 38 are adequately addressed for the safe shutdown systems.

The staff evaluated whether 10 CFR 50.55a(h) is adequately addressed for the safe shutdown systems. The regulation at 10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995, which is evaluated in Sections 7.4.3.1 and 7.4.3.2 of this report and finds it to be adequately addressed. Accordingly, the staff finds that 10 CFR 50.55a(h) is adequately addressed for the for the safe shutdown systems.

The staff evaluated whether the applicant meets the requirements of 10 CFR 52.47(b)(1). This regulation requires that the application for design certification must contain 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. Section 7.2.3 of this report addresses the ITAAC specific to the safe shutdown systems. The staff evaluation of conformance to 10 CFR 52.47 in Section 7.1.1.3.4 of this report applies to the safe shutdown systems. Therefore, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the safe shutdown systems.

7.4.4 Conclusion

Based on the above, the staff concludes that the applicant adequately addresses the major design considerations for the safe shutdown systems. As discussed in Sections 7.1.1.3.1 through 7.1.1.3.10 of this report and Section 7.4.3 above, the staff concludes that, for the safe shutdown systems, the applicant adequately addresses the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 52.47(b)(1); and GDC 1, 2, 4, 13, 19, 24, 34, 35, and 38. The applicant also identified adequate high-level functions and included sufficient DAC/ITAAC in DCD Tier 1, Revision 9, to verify that the design of the safe shutdown systems is completed in compliance with the applicable requirements.

7.5 Information Systems Important to Safety

7.5.1 Introduction

The staff reviewed the information systems important to safety in accordance with SRP Section 7.5, Revision 5, to confirm that these systems will provide the information necessary to ensure plant safety during all plant conditions for which they are required.

SRP Section 7.5 provides acceptance criteria for the following types of systems:

- Accident monitoring instrumentation
- Bypassed or inoperable status indication (BISI) for safety systems
- Plant annunciator (alarm) systems
- SPDS and information systems associated with the emergency response facilities (ERF) and ERDS

7.5.1.1 *Summary of Regulatory Criteria*

Specific acceptance criteria are identified for each type of system. Accordingly, acceptance criteria are identified that apply to all information systems important to safety, followed by the additional acceptance criteria particular to each of the types of information systems important to safety, as described below:

- Acceptance criteria for all information systems important to safety:

    Acceptance criteria for all information systems important to safety are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1 and 24; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152.

- Additional acceptance criteria for accident monitoring instrumentation:

  In addition to the requirements listed above for the information systems important to safety, acceptance criteria for the accident monitoring instrumentation are based on meeting the relevant requirements of 10 CFR 50.34(f)(2), Subparts (v) [I.D.3], (xi) [II.D.3], (xvii) [II.F.1], (xviii) [II.F.2], (xix) [II.F.3], and (xxiv) [II.K.3.23], as well as GDC 2, 4, 13, and 19.

- Additional acceptance criteria for BISI for safety systems:

  In addition to the requirements for the information systems important to safety, acceptance criteria for the BISI are based on meeting the relevant requirements of 10 CFR 50.34(f)(2)(v) [I.D.3].

- Additional acceptance criteria for plant annunciator (alarm) systems:

  In addition to the requirements for the information systems important to safety, acceptance criteria for the annunciator systems are based on meeting the relevant requirements of GDC 13 and 19, and the SRM to SECY-93-087, Item II.T.

SRP Section 7.5 does not identify additional requirements for SPDS, ERF information systems, and ERDS information systems.

In addition to using SRP Section 7.5, the staff reviewed I&C systems with accident monitoring functions in accordance with BTP HICB-10, "Guidance on Application of Regulatory Guide 1.97." RG 1.97 describes methods acceptable to the staff for providing instrumentation to monitor variables for accident conditions. BTP HICB-10 requires use of the regulatory criteria in 10 CFR 50.34(f)(2)(xvii) and GDC 13, 19, and 64, "Monitoring Radioactivity Releases." Note that the acceptance criteria in BTP HICB-10 are redundant to the criteria in SRP Section 7.5, with the exception of GDC 64.

The applicant identified a CMS as an information system important to safety. In addition to the requirements for the accident monitoring instrumentation, acceptance criteria for the CMS are based on meeting the relevant requirements of 10 CFR 50.44(c)(4).

7.5.1.2 *Method of Review*

As noted above, SRP Section 7.5 provides acceptance criteria for four types of systems. DCD Tier 2, Revision 9, Section 7.5, directly describes one of the systems, accident monitoring instrumentation. For the remaining three types of systems, DCD Tier 2, Revision 9, Section 7.5, mentions them briefly and identifies where they are discussed in greater detail in the DCD. This report evaluates each of these systems.

In addition to the PAM instrumentation, DCD Tier 2, Revision 9, identifies four information systems (CMS, PRMS, ARMS, and pools monitoring system) that are not directly covered by SRP Section 7.5. For the CMS, the staff used the criteria for accident monitoring instrumentation, with the exception of criteria related to RG 1.97, since these criteria are addressed for the PAM instrumentation. The staff also identifies where the remaining three systems are evaluated in this report.

Since the DCD and the SRP do not fully match, the evaluation of information systems important to safety is evaluated in a hybrid manner. For each applicable system, the staff provides a summary of technical information and an evaluation of major design considerations. The staff then provides a general evaluation of the regulatory criteria.

7.5.2 Post Accident Monitoring Instrumentation

7.5.2.1 *Regulatory Criteria*

Acceptance criteria for the accident monitoring instrumentation are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2), Subparts (v) [I.D.3], (xi) [II.D.3], (xvii) [II.F.1], (xviii) [II.F.2], (xix) [II.F.3], and (xxiv) [II.K.3.23]; GDC 1, 2, 4, 13, 19, 24, and 64; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152.

7.5.2.2 *Summary of Technical Information*

The safety portion of the PAM systems consists of those systems that provide information for the safe operation of the plant during normal operation, AOOs, and accidents to help ensure performance of manual safety functions. The safety information systems include those systems that provide information for manual initiation and control of safety systems, indicate that safety plant functions are being accomplished, and provide information from which appropriate actions can be taken to mitigate the consequences of accidents.

The nonsafety portion of the PAM systems includes the SPDS, information systems associated with the ERF, and the ERDS, none of which performs safety functions.

RG 1.97 endorses (with certain exceptions specified in Section C of the RG) IEEE Std 497, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." IEEE Std 497 establishes flexible, performance-based criteria for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables. IEEE Std 497 identifies five types of variables for accident monitoring and the criteria for the selection of each type of variable.

The PAM instrumentation design is part of the overall HFE process. The HFE process includes the functional requirements analysis, allocation of functions, and task analysis that address critical safety functions and provides an independent list of the required RG 1.97 parameters via their respective results summary reports. The functional requirements analysis (FRA), allocation of functions (AOF), and task analysis (TA) are iteratively integrated into the design process to provide a final design that effectively balances human factors and system design. The list of parameters, generated by the HFE process, is compared with the information generated from the design process and the differences are entered into the HFE issue tracking system for resolution.

The PAM variable list is prepared as a separate document, using inputs from the design process, licensing design basis, and HFE process, including the development of the emergency procedure guidelines and/or emergency operating procedures and abnormal operating procedures. The PAM variable list document provides summary information for each PAM variable as applicable. Typical information provided includes the following:

- PAM variable name
- Type
- Range

- Extended range (Type C)
- Instrument channel accuracy
- Required instrument duration
- Power source
- Required number of channels
- Qualification criteria
- Type of monitoring channel display

In DCD Tier 1, Revision 9, Section 3.7, the applicant documented the DAC for the PAM instrumentation as follows:

<u>Performance Criteria</u>

- Range
- Accuracy
- Response time
- Required instrument duration
- Reliability
- Performance assessment documentation

<u>Design Criteria</u>

- Single failure
- CCFs
- Independence and separation
- Isolation
- Information ambiguity
- Power supply
- Calibration
- Testability
- Direct measurement
- Control of access
- Maintenance and repair
- Minimizing measurements
- Auxiliary supporting features
- Portable instruments
- Documentation of design criteria

<u>Qualification Criteria</u>

- Type A variables
- Type B variables
- Type C variables
- Type D variables
- Type E variables
- Portable instruments
- Post-event operating time
- Documentation of qualification criteria

Display Criteria

- Information characteristics
- Human factors
- Anomalous indications
- Continuous versus on-demand display
- Trend or rate information
- Display identification
- Type of monitoring channel display
- Display location
- Information ambiguity
- Recording
- Digital display signal validation
- Display criteria documentation

In DCD Tier 1, Revision 9, Section 3.7, the applicant documented that the ITAAC will be performed using the FRA, AOF, and TA to support closure of the referenced ITAAC that will provide the final list of the required RG 1.97 parameters, via their respective results summary reports, to verify that the PAM instrumentation is installed consistent with the selected variables as described above.

7.5.2.3 *Evaluation of Accident Monitoring Systems Conformance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.5, the following are the major design considerations that should be emphasized in the accident monitoring systems review.

(1) Conformance with RG 1.97 and BTP HICB-10

The staff evaluated whether the guidelines of RG 1.97 are adequately addressed. RG 1.97 endorses IEEE Std 497. DCD Tier 2, Revision 9, Section 7.5.1.3.4, states that the ESBWR conforms to RG 1.97 and IEEE Std 497 (with certain exceptions specified in Section C of RG 1.97). DCD Tier 2, Section 7.5.1.3, also describes the performance-based criteria that the ESBWR uses for the selection, performance, design, qualification, display, and quality assurance of accident monitoring variables. These criteria are consistent with RG 1.97 and IEEE Std 497. Conformance with IEEE Std 497, Sections 6.2, and 8 is described under Item (2) below. The selection of accident monitoring variables is integrated with the HFE design, as described in DCD Chapter 18. DCD Tier 1, Revision 9, Section 3.7, includes these criteria and ITAAC to confirm that the PAM instrumentation is installed and consistent with the selected variables. DCD Tier 1, Revision 9, Section 3.8, includes the DAC/ITAAC to confirm that the HFE design is implemented in accordance with the process described in DCD Chapter 18. Accordingly, the staff finds that the guidelines of RG 1.97 are adequately addressed.

The staff evaluated whether the guidelines of BTP HICB-10 are adequately addressed. BTP HICB-10 includes acceptance criteria that supplement the design and qualification criteria identified in RG 1.97, Revisions 2, 3, and 4. These include (1) EQ, (2) seismic qualification, (3) redundancy, (4) independence of redundant instrumentation, (5) display and recording, (6) range, (7) minimizing measurements, (8) alternate variables, (9) guidance for BWR and PWR variables, (10) conversion to Revision 4, and (11) modifications to Revision 4. However, several of these considerations apply only to current plants using RG 1.97, Revisions 2 or 3. Because the applicant is applying RG 1.97, Revision 4, the only criteria that apply are EQ, seismic qualification, independence of redundant instrumentation, range, and minimizing measurements. Each of these considerations is evaluated below.

DCD Tier 2, Revision 9, Section 7.5.1.3.5, discusses conformance of the PAM instrumentation to BTP HICB-10. The section references RG 1.97, Revision 4, Section A, which states, "Branch Technical Position HICB 10 will require updates for consistency with Revision 4 of RG 1.97. Conformance to these requirements is addressed during the detailed design phase." In RAI 7.5-7, the staff requested the applicant to clarify conformance to RG 1.97 and BTP HICB-10. RAI 7.5-7 was being tracked as an open item in the SER with open items. In its response, the applicant clarified the PAM design basis and made corresponding changes to DCD Revision 6. DCD Tier 1, Revision 9, Section 3.7, committed that the installed PAM instrumentation (scope as determined by the HFE process as described in DCD Tier 1, Revision 9, Section 3.3) conforms to the requirements (variables, types, performance criteria, design criteria, qualification criteria, display criteria, and quality assurance), as outlined in RG 1.97. Accordingly, the staff finds that the PAM instrumentation design follows the guidelines of RG 1.97 and BTP HICB-10. The staff finds that the response is acceptable since the applicant addressed conformance to RG 1.97 and BTP HICB-10. Based on the applicant's response, RAI 7.5-7 is resolved.

BTP HICB-10 also identifies GDC 64 as part of the regulatory basis, which requires, in part, that means be provided to monitor (1) the reactor containment atmosphere, (2) spaces containing components for recirculation of LOCA fluid, (3) effluent discharge paths, and (4) the plant environs for radioactivity that may be released from postulated accidents. DCD Tier 2, Revision 9, Section 7.5.1.3.2, states that GDC 64 applies to the PAM instrumentation. DCD Tier 2, Revision 9, Section 7.3.3, describes the LD&IS, which provides monitoring radioactivity inside and outside containment. The LD&IS also receives information from the RCPB leak detection systems, as described in DCD Revision 9, Section 5.2.5. DCD Tier 2, Revision 9, Sections 7.5.3 and 11.5, describe the PRMS, which provides a capability for determining the content of radioactive material in various gaseous and liquid process and effluent streams. DCD Tier 2, Revision 9, Section 11.5.5.4, specifically identifies ESBWR areas that are monitored in conformance with GDC 64 that include spaces containing components for recirculation of LOCA fluid. Information from the LD&IS and PRMS is available to the PAM instrumentation through the DCIS. DCD Tier 2, Revision 9, Section 7.5.1.3, describes the criteria that the ESBWR uses for the selection of accident monitoring variables. In the ESBWR design, the LOCA fluid is passively recirculated through the GDCS, which receives condensed steam from the PCCS. Since both the GDCS and PCCS are inside containment, the monitoring is provided by the LD&IS. The selection of accident monitoring variables is integrated with the HFE design process, as described in DCD Chapter 18. DCD Tier 1, Revision 9, Section 3.7, includes these criteria and ITAAC to confirm that the PAM instrumentation is described consistently with the selected variables. DCD Tier 1, Revision 9, Section 3.8, includes the DAC/ITAAC to confirm that the HFE design is implemented according to the process described in DCD, Revision 9, Chapter 18. Accordingly, the staff finds that the requirements of GDC 64 are adequately addressed.

(2) Use of Digital Systems (IEEE Std 497-2002, Sections 6.2 and 8)

SRP Section 7.5 identifies that the review of computer-based digital systems should focus on IEEE Std 497, Sections 6.2 and 8. IEEE Std 497-2002, Section 6.2, "Common Cause Failure," states that the design should address the concern of CCFs of the digital system. The applicant submitted NEDO-33251 to demonstrate that defense-in-depth exists against the consequences of a software CCF. Section 7.1.3 of this report documents the staff's evaluation of the D3 assessment.

IEEE Std 497, Section 6.2, states that use of identical software in redundant instrumentation channels is acceptable, provided that the licensee conducts an analysis to demonstrate D3 exists against CCFs. For accident monitoring instrumentation from the safety sources, the Q-DCIS provides the required signal path to process this information. This information then is shown on the Q-DCIS divisional safety displays. The safety information can also be transmitted via isolated safety gateways to the N-DCIS for input to nonsafety displays, PCF, and AMS. Type A, Type B, and Type C variables are powered from safety sources. For Type D and Type E variables, which are powered from nonsafety sources, the N-DCIS provides the required signal paths to process information. As discussed in Section 7.1 of this report, the DCIS design satisfies the separation and isolation guidelines. The staff finds this arrangement acceptable.

IEEE Std 497, Section 8, specifies the "display criteria" for accident monitoring variables that should include the results of an analysis of the system functions required to respond to an accident and analysis of the tasks required of the operator to implement those functions during DBEs. Display characteristics should be identified that include, as a minimum, range, instrument accuracy, precision, display format (e.g., status, value, or trend), units, and response time.

DCD Tier 1, Revision 9, Section 3.7, documents the design requirements for the ESBWR PAM instrumentation. The DCIS (both the Q-DCIS and the N-DCIS) provides the required signal paths to process this information. For variables associated with critical safety functions and powered from a safety power source, the Q-DCIS provides the required signal paths to process this information. The information is then displayed on the Q-DCIS divisional safety displays. The safety information can also be transmitted via isolated nonsafety gateways to the N-DCIS for input to nonsafety displays, PCF, and the AMS. Type A, Type B, and Type C variables are powered from safety sources. Type D and Type E variables will have their power source determined as part of the design process.

DCD Tier 2, Revision 9, Section 7.5.1.3, describes the criteria that the ESBWR uses for the selection of accident monitoring variables. The selection of accident monitoring variables is integrated with the HFE design process, as described in DCD Tier 2, Revision 9, Chapter 18. DCD Tier 1, Revision 9, Section 3.7, includes these criteria and the ITAAC to confirm that the PAM instrumentation is installed consistently with the selected variables. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC to confirm that the HFE design is implemented in accordance with the process described in DCD Chapter 18. Accordingly, the staff finds that the requirements of IEEE Std 497, Section 8, are adequately addressed.

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed for the PAM instrumentation. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2 concerning the use of digital systems. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in SRP Appendix 7.1-D. The staff's evaluation of conformance to IEEE Std 7-4.3.2 in Section 7.1.1.3.10 of this report applies to the PAM system.

NEDE-33226P describes the software development activities and NEDE-33245P describes the software QA activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in

NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff's evaluation of software development and software QA activities.

(3) Emergency Operating Procedure Action Points

SRP Section 7.5 states that a basis should be provided for the EOP action points that account for measurement uncertainties. EOP action points are type B accident monitoring variables under RG 1.97, Revision 4. DCD Tier 2, Revision 9, Section 7.5.1.3, describes the performance criteria and design criteria to be used in accident monitoring instrumentation, including instrumentation for type B accident monitoring variables. The performance criteria and design criteria include considerations for variable accuracy, information ambiguity, and calibration. DCD Tier 1, Revision 9, Section 3.7, provides ITAAC to confirm that the accident monitoring instrumentation meets the performance and design criteria of RG 1.97. Based on the above, the staff finds that the EOP action points and their measurement uncertainties are adequately addressed.

(4) Monitoring for Severe Accidents

SRP Section 7.5 states the following:

> The accident monitoring instrumentation should be demonstrated to perform their intended function for severe accident protection. They need not be subject to additional 10 CFR 50.49 environmental qualification requirements. However, they should be designed so that there is reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed.

This guidance is based on the SRM to SECY-93-087, "Item L Equipment Survivability."

DCD Tier 2, Revision 9, Section 19.3.4, summarizes the ESBWR severe accident equipment survivability analysis. Appendix 8D to NEDO-33201, "ESBWR Certification Probabilistic Risk Assessment," provides a more detailed severe accident equipment survivability analysis. NEDO-33201, Table 8.D.2-1, identifies the required functions and associated monitored variables. NEDO-33201, Section 8D.4.6, provides the equipment capability evaluation of the PAM equipment. Section 19.2.3.3.8 of this report provides the staff's evaluation of the severe accident equipment survivability analysis and finds that the analysis provides reasonable assurance that the equipment necessary to achieve a controlled, stable plant condition will function over the time span in which it is needed. Accordingly, the staff finds that the guidance on monitoring for severe accidents is adequately addressed.

(5) Performance Assessment

SRP Section 7.5 identifies that the review should confirm that the performance assessment fulfills the goals outlined in IEEE Std 497-2002, Section 5.6, "Performance Assessment Documentation," which states that an assessment for each of the performance criteria shall be conducted to assure that the as-designed performance meets or exceeds the performance criteria. DCD Tier 2, Revision 9, Section 7.5.1.3.4, states that performance criteria (identified in IEEE Std 497, Section 5) are developed during the design process using input from the HFE process together with other design and accident analysis inputs. The PAM variable list documents the performance criteria for each required variable. Performance is verified to meet the as-designed performance criteria of DCD Tier 2, Revision 9, Section 18.11. DCD Tier 1, Revision 9, Section 3.7, includes these performance criteria and ITAAC to confirm that the PAM instrumentation is installed consistent with the selected variables. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC to confirm that the HFE design is implemented in accordance with the process described in DCD Tier 2, Revision 9, Chapter 18. Accordingly, the staff finds that the guidance on performance assessment is adequately addressed.

The remaining criteria are evaluated in Section 7.5.11 of this report.

7.5.3 Containment Monitoring System

7.5.3.1 *Regulatory Criteria*

The CMS is classified as a safety and seismic Category I system. Acceptance criteria for the CMS are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2), Subparts (v) [I.D.3], (xi) [II.D.3], (xvii) [II.F.1], (xviii) [II.F.2], (xix) [II.F.3], and (xxiv) [II.K.3.23]; 10 CFR 50.44(c)(4); GDC 1, 2, 4, 13, 19, and 24; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152.

7.5.3.2 *Summary of Technical Information*

The CMS provides the instrumentation to monitor the following:

- Atmosphere in the containment for high gross gamma radiation levels
- Pressure of the drywell and wetwell
- Drywell and wetwell differential pressure
- Lower and upper drywell water level (post-LOCA)
- Temperature of the suppression pool water
- Suppression pool water level
- Drywell and wetwell hydrogen and oxygen concentration
- Containment area radiation

These parameters are monitored during both normal reactor operations and post accident conditions to evaluate the integrity and safe conditions of the containment. Abnormal measurements and indications initiate alarms in the MCR.

The CMS is divisional and segregated (safety/nonsafety). The specific system features are as follows:

- Radiation monitoring and hydrogen and oxygen sampling are provided for the drywell and for the air space above the suppression pool.

- Each radiation monitoring channel uses one gamma-sensitive ion chamber and one digital log radiation monitor. Four channels are provided, two for the drywell and two for the suppression pool (wetwell) air space.

- During normal plant operation, both the radiation monitoring and gas sampling subsystems are operating. For PAM, the gas sampling subsystem is automatically activated by the LOCA signal to alternate its sampling between the drywell and the wetwell. The area of sampling can be selected manually or sequentially controlled.

- Heat tracing is provided on the gas sampling lines for control of moisture and condensation.

- Two isolation valves are provided on each sample and return line that penetrates the containment. Each line has one manual inner valve and one remote-control outer valve.

- Each gas sampling analyzer has dual redundant pumps. One is used during normal operation; the other is used for added capacity or backup.

- Separate oxygen and hydrogen gas sources are provided in each CMS sampling rack with known compositions for monitor calibration.

- CMS piping connections are provided.

- The drywell pressure instrumentation taps are located throughout the containment, and the sensors are located outside the containment.

- Four drywell pressure transmitters are provided for safety signals for use by the RPS for reactor scram. Four additional safety drywell pressure signals are made available to the LD&IS, where they are used to initiate isolation of containment valves, transfer pump suction, and initiate SPCS.

- Four drywell water level sensors are provided as safety-related signals for use by the LD&IS for feedwater line isolation and FW ASD controller breaker trip.

- Two wide-range safety pressure transmitters are used for providing safety drywell pressure information meeting the requirements of PAM.

- Four nonsafety drywell pressure transmitters are used by the DPS for diverse scram protection monitoring and by the CIS for controlling the position of the nitrogen makeup pressure control valve.

- The suppression pool water level is monitored during all plant operating conditions and post accident conditions. Suppression pool water level monitoring consists of 10 channels of water level detection sensors distributed into four safety narrow-range and four nonsafety wide-range instruments. The narrow-range suppression pool water level signals are used to detect the uncovering of the first set of suppression pool temperature sensors below the pool surface. When the suppression pool water level drops below the elevation of a particular set of temperature sensors, those sensor signals are not used in computing the average pool temperature.

- Two of the wide-range water level signals are used for displaying suppression pool water level on the RSS panels.

- Suppression pool temperatures are monitored.

7.5.3.3 *Staff Evaluation*

Section 7.2 of this report evaluates the SPTM function of the CMS, which is part of the RPS.

The staff evaluated whether the I&C portions of 10 CFR 50.44(c)(4) are adequately addressed. 10 CFR 50.44(c)(4)(i) requires that equipment be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond DBA for combustible gas control and accident management, including emergency planning. In addition, 10 CFR 50.44(c)(4)(ii) requires that equipment be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond DBA for accident management, including emergency planning.

DCD Tier 2, Revision 9, Section 6.2.5, describes the design of the oxygen and hydrogen monitors, which is evaluated in Section 6.2.5 of this report. DCD Tier 2, Revision 9, Section 7.5.2.3.1 specifies that the CMS conforms to 10 CFR 50.44(c)(4). DCD Tier 2, Section 7.5.2.3 identifies that the CMS provides continuous monitoring during normal reactor operation, as well as during and after DBEs. DCD Tier 2, Revision 9, Section 7.5.2.1, indicates that the safety hydrogen/oxygen analyzers are active during normal operation. Additional sampling capacity is automatically initiated by a LOCA signal for PAM of oxygen and hydrogen content in the containment. DCD Tier 2, Revision 9, Section 7.5.2.5, describes the surveillance testing of the CMS, which includes instrument channel checks of the radiation and gas monitors, functional tests to verify equipment operability, sensor calibration and response tests, and leakage tests of the gas sampling lines. DCD Tier 2, Revision 9, Table 7.5-5, identifies the instrument ranges for hydrogen and oxygen analyzers. The staff finds that these design bases address the functionality, reliability, and capability requirements in 10 CFR 50.44(c)(4). Accordingly, the staff finds that the requirements of 10 CFR 50.44(c)(4) are adequately addressed for the CMS.

Section 7.5.11 of this report evaluates the remaining criteria.

7.5.4 Process Radiation Monitoring System

The PRMS provides the instrumentation for radiological monitoring, sampling, and analysis in the following areas:

- Turbine building
- TSC
- Radwaste building
- Control building
- Reactor building
- Fuel building
- Reactor building and fuel building stack
- Turbine building stack
- Radwaste building stack

The PRMS alerts operators to radiation levels in excess of preset limits and initiates automatically the required protection action to isolate, contain, or redirect radioactivity releases to the environs. Some subsystems of the PRMS are safety. Section 11.5 of this report provides the evaluation of the PRMS.

7.5.5 Area Radiation Monitoring System

The primary function of the nonsafety ARMS is to continuously monitor the gamma radiation levels within the various areas of the plant and to provide an early warning that predetermined radiation levels are exceeded. The ARMS consists of area radiation detectors located at accessible areas of the plant and utilizes local and MCR alarms for immediate warning. The gross gamma radiation levels are monitored on a continuous basis because changes are caused by operational transients or maintenance activities. Any high radiation levels are indicated by audible area alarms and MCR alarms. Section 12.4.3.4 of this report addresses the evaluation of these systems.

7.5.6 Pool Monitoring Subsystem Evaluation

The CMS provides safety temperature and level instrumentation to monitor suppression pool water temperature and water level, respectively. Section 7.5.3 of this report evaluates the CMS.

The GDCS provides safety level instrumentation for the GDCS pools to provide necessary information to the operator for maintaining the GDCS water level required for the safety ECCS function. Section 7.3 of this report evaluates the instrumentation for the GDCS.

Safety level instrumentation is provided in the FAPCS for the spent fuel pool, the buffer pool, and the IC/PCCS pools to detect a low water level that would indicate a loss of decay heat removal ability. Section 9.1.3 of this report evaluates the FAPCS.

7.5.7 Not Used

7.5.8 Bypassed and Inoperable Status Indication for Safety Systems

7.5.8.1 *Regulatory Criteria*

Acceptance criteria for BISI for safety systems are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(v) [I.D.3]; GDC 1 and 24; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152.

7.5.8.2 *Summary of Technical Information*

DCD Tier 2 does not directly address BISI for safety systems consistent with SRP Section 7.5. DCD Tier 2, Revision 9, Section 7.5, mentions BISI for safety systems briefly and identifies where they are discussed in greater detail in the DCD. DCD Tier 2, Revision 9, Table 1A-1, discusses BISI for safety systems in the context of conformance to 10 CFR 50.34(f)(2)(v) and identifies where information is provided in more detail in DCD Tier 2, Revision 9, Sections 7.2, 7.3, 7.5, and 7.8. BISI for safety systems is also discussed in the context of conformance to IEEE Std 603, Section 5.8.3, and RG 1.47 in associated sections.

7.5.8.3 *Evaluation of BISI for Safety Systems Conformance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.5, the following major design considerations should be emphasized in the BISI for safety systems review:

(1) Scope of Bypassed and Inoperable Status Indication

SRP Section 7.5 notes that, at a minimum, BISI should be provided for four sets of systems; however, only the first sets of systems, the RPS and ESF actuation systems, apply to the design. DCD Tier 2, Revision 9, Table 7.1-1, states that 10 CFR 50.34(f)(2)(v), which is the requirement to provide for automatic indication of the bypassed and inoperable status of safety systems, applies to all safety systems, including the RPS and ESF. Section 7.1.1.3.4 of this report evaluates the I&C systems against the requirements of 10 CFR 50.34(f)(2)(v). SRP Section 7.5 notes that the indication of bypasses should conform to IEEE Std 603, Section 5.8.3.

As described in Sections 7.1.1.3.4 (for 10 CFR 50.34(f)(2)(v)) and 7.1.1.3.10 (for IEEE Std 603, Section 5.8.3) of this report, the staff finds that IEEE Std 603, Section 5.8.3, is adequately addressed based on its inclusion in the safety systems design bases and DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria. Information displays are designed using the HFE design process, as described in DCD Tier 2, Revision 9, Chapter 18, and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for verifying the implementation of the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. This verification applies to all safety systems and includes BISI. Accordingly, based on the inclusion of applicable criteria in the safety systems design basis and their confirmation in the DAC/ITAAC, the staff finds that the guideline for the scope of BISI is adequately addressed.

(2) Conformance with Regulatory Guide 1.47

RG 1.47 provides more specific guidance on BISI. DCD Tier 2, Revision 9, Table 7.1-1, indicates that RG 1.47 applies to the safety systems consistent with SRP Table 7-1. DCD Tier 2, Revision 9, Section 7.1.6.4, in the discussion regarding conformance to RG 1.47, notes that bypass indications are designed to satisfy the guidance of IEEE Std 603, Section 5.8.3, and RG 1.47. This section also states that bypass indications use isolation devices that preclude the possibility of any adverse electrical effect of the bypass indication circuits on the plant safety system. The staff finds this acceptable.

(3) Independence (IEEE Std 603, Sections 5.6 and 6.3)

SRP Section 7.5 states that the BISI for safety systems should be designed and installed in a manner that precludes the possibility of adverse effects on plant safety systems, in conformance with IEEE Std 603, Sections 5.6 and 6.3. As described in Section 7.1.1.3.10 of this report, which applies to BISI for safety systems, the staff finds that IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed based on their inclusion in the safety systems design basis and their confirmation in the DAC/ITAAC. Accordingly, the staff finds IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed for BISI.

(4) Use of Digital Systems

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed for BISI. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2 concerning the use of digital systems. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in SRP Appendix 7.1-D. The staff evaluation of conformance to IEEE Std 7-4.3.2 in Section 7.1.1.3.10 of this report applies to the BISI system.

NEDE-33226P describes the software development activities and NEDE-33245P describes the software QA activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff's evaluation of software development and software QA activities.

Section 7.5.11 of this report evaluates the remaining criteria.

7.5.9 Plant Annunciator (Alarm) Systems

7.5.9.1 *Regulatory Criteria*

SRP Section 7.5 acceptance criteria for information systems important to safety are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1, 13, 19, and 24; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152, and the SRM to SECY-93-087, Item II.T.

7.5.9.2 *Summary of Technical Information*

DCD Tier 2 does not directly address annunciator systems consistent with SRP Section 7.5. Annunciator systems are described in the context of the AMS and its conformance to the SRM to SECY-93-087, Item II.T. DCD Tier 2, Revision 9, Sections 7.1.5.3 and 7.1.6.3 state that the AMS has the following attributes:

- The AMS follows the guidance in SECY-93-087, Item II.T, including following the guidance for redundancy, independence, and separation.
- Alarm points are sent through dual networks to redundant message processors on dual power supplies.
- The processors are dedicated to only performing alarm processing.
- The alarms are displayed on multiple independent VDUs, each of which has dual power supplies.
- The alarm tiles, or their equivalent, are driven by redundant data links (with dual power).
- There are redundant alarm processors.
- No alarms require manually controlled actions for safety systems to accomplish their function.
- There is one horn and one voice speaker.
- Test buttons test the horn and the lights.
- The requirements for safety equipment and circuits are not applicable.

7.5.9.3 *Evaluation of Plant Annunciator Systems Conformance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.5, the following major design considerations should be emphasized in the plant annunciator systems review:

(1) Reliability (IEEE Std 603, Section 5.15)

SRP Section 7.5 states that the applicant should justify that the degree of redundancy, diversity, testability, and quality provided in annunciator systems is adequate to support normal and emergency operations. DCD Tier 2, Revision 9, Section 7.1.5.3, notes that the AMS conforms to the SRM to SECY-93-087, Item II.T, including following the guidance for redundancy, independence, and separation. The staff evaluated conformance to the SRM to SECY-93-087, Item II.T, in Section 7.1.1.3.7 of this report and finds it acceptable. Since the AMS is a nonsafety system, it is not required to conform to the IEEE Std 603 criteria. Therefore, the staff performed a general review of the AMS. The staff finds that the AMS attributes identified in Section 7.5.9.2 of this report, in combination with the implementation of the digital system guidelines and self-test provisions in Items (2) and (5) below, support adequate redundancy, diversity, testability, and quality.

DCD Tier 2, Revision 9, Section 7.1.6.3, indicates that no alarms require manually controlled actions for safety systems to accomplish their function. Accordingly, the staff concurs with the applicant that the requirements for manually controlled safety equipment and circuits are not applicable.

DCD Tier 2, Revision 9, Section 7.1.5.4, describes the testability of the N-DCIS, which includes the annunciator systems. The N-DCIS controllers are equipped with online diagnostic capabilities for cyclically monitoring the operability of input/output (I/O) signals, buses, power supplies, processors, and interprocessor communications and that the online diagnostics are performed without interrupting the normal operation of the N-DCIS. The staff finds these testability features acceptable for nonsafety systems since online diagnostic equipment is routinely applied for such applications.

The applicant has described an integrated development process that is confirmed by ITAAC to ensure that the annunciator systems have the necessary reliability and functionality. The MCR, including the annunciator systems, is designed using the HFE design process, as described in DCD Tier 2, Revision 9, Chapter 18, and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. The software for the MCR, including the annunciator systems, is developed using the software development activities described in NEDE-33226P and its quality assessed and assured by the software QA activities as described in NEDE-33245P. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2.3.7 of this report provides the staff evaluation of the development process for software for nonsafety systems, which includes the annunciator systems. Based on the above, the staff finds that the reliability guidelines are adequately addressed.

(2) Use of Digital Systems

NEDE-33226P describes the software development activities and NEDE-33245P describes the software QA activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff's evaluation of software development and software QA activities. As described in Item (1) above, these software development activities apply to the MCR and its alarms. Accordingly, based on the applicability of the software development activities to the annunciator system and the confirmation of their implementation in the DAC/ITAAC, the staff finds that the use of digital systems guidelines is adequately addressed.

(3) Independence (IEEE Std 603, Sections 5.6 and 6.3)

SRP Section 7.5 notes that the annunciator systems should be evaluated for isolation between safety systems and other systems in conformance with IEEE Std 603, Sections 5.6 and 6.3. As described in Section 7.1.1.3.10 of this report, which applies to the annunciator systems, the staff finds that IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed based on their inclusion in the safety systems design basis and their confirmation in the DAC/ITAAC. Accordingly, the staff finds that IEEE Std 603, Sections 5.6 and 6.3, are adequately addressed for annunciator systems.

(4) Redundancy

As described in Item (1) above, DCD Tier 2, Revision 9, Section 7.1.5.3, indicates that the AMS conforms to the SRM to SECY-93-087, Item II.T, including following the guidance for redundancy, independence, and separation. The staff evaluated conformance to the SRM to SECY-93-087, Item II.T, in Section 7.1.1.3.7 of this report and finds it acceptable. The AMS has several redundant features, including (1) sending alarm points through dual networks to redundant message processors on dual power supplies, (2) displaying alarms on multiple independent VDUs that each have dual power supplies, (3) driving alarm tiles, or their equivalent, by redundant datalinks (with dual power), and (4) using redundant alarm processors. The MCR, including the alarms, is designed using the HFE design process, as described in DCD Tier 2, Revision 9, Chapter 18, and evaluated in Chapter 18 of this report. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing the HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. Based on the above, the staff finds that the redundancy guidelines are adequately addressed.

(5) Self-Test Provisions (BTP HICB-17)

BTP HICB-17 provides self-test provisions for safety and protections. Provisions applicable to the nonsafety annunciator systems include self-test features and actions upon failure detection. DCD Tier 2, Revision 9, Section 7.1.5.4, notes that the N-DCIS controllers are equipped with online diagnostic capabilities to identify and isolate failure of I/O signals, buses, power supplies, processors, and interprocessor communications and that these online diagnostics can be performed without interrupting the normal operation of the N-DCIS. Accordingly, the staff finds that the redundancy self-test provisions are adequately addressed.

(6) Compliance with IEEE Std 603[1991]

SRP Section 7.5 indicates that IEEE Std 603 applies when alarms are provided for manually controlled actions that are required for the safety systems to accomplish their safety functions and for which no automatic control is provided. As described in Item (1) above, this is not applicable to the design. The annunciator systems are nonsafety and therefore must be isolated from the safety systems. As described in Item (3) above and Section 7.1.1.3.10 of this report, which applies to the annunciator systems, the staff finds that Sections 5.6 and 6.3 of IEEE Std 603 are adequately addressed based on their inclusion in the safety systems design basis and their confirmation in the DAC/ITAAC. Accordingly, the staff finds that compliance with IEEE Std 603 is adequately addressed for the annunciator systems.

Section 7.5.11 of this report evaluates the remaining criteria.

7.5.10 Safety Parameter Display System, Emergency Response Facilities Information Systems, and Emergency Response Data System Information Systems

7.5.10.1 *Regulatory Criteria*

Acceptance criteria for information systems important to safety are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1 and 24; and 10 CFR 52.47(b)(1). The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152.

7.5.10.2 *Summary of Technical Information*

DCD Tier 2, Revision 9, Section 7.1.5.1.2 indicates that the SPDS, the ERF and the ERDS information systems are nonsafety and portions of the N-DCIS.

7.5.10.3 *Evaluation of the Safety Parameter Display System and the Emergency Response Facility and Emergency Response Data System Information Systems Conformance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.5, the following major design considerations should be emphasized in the SPDS and the ERF and ERDS information systems review:

(1) Independence (IEEE Std 603, Sections 5.6 and 6.3)

DCD Tier 2, Revision 9, Section 7.1.4.2, states that the SPDS and the ERF and ERDS information systems are nonsafety and portions of the N-DCIS. SRP Section 7.5 notes that, for the SPDS and the ERF and ERDS information systems isolated from the protection system, the applicable requirements of 10 CFR 50.55a(h) for IEEE Std 603 are Sections 5.6.3 and 6.3. DCD Tier 2, Revision 9, Sections 7.1.6.6.1.7 and 7.1.6.6.1.19, both indicate that the safety systems are separated and independent from nonsafety portions of systems in conformance with IEEE Std 603, Sections 5.6.3 and 6.3. Section 7.1.1.3.10 of this report provides an evaluation of IEEE Std 603, Sections 5.6.3 and 6.3, which applies to the SPDS and the ERF and ERDS information systems. In Section 7.1.1.3.10 of this report, the staff finds that Sections 5.6.3 and 6.3 of IEEE Std 603 are adequately addressed based on their inclusion in the safety systems' design basis and their confirmation in the DAC/ITAAC. Accordingly, the staff finds that Sections 5.6.3 and 6.3 in IEEE Std 603 are adequately addressed for the SPDS and the ERF and ERDS information systems.

Section 7.5.11 of this report evaluates the remaining criteria.

7.5.11 Evaluation of Information Systems Important to Safety Common Acceptance Criteria

The staff reviewed the applicable regulations for the information systems important to safety in accordance with SRP Sections 7.5 and 7.1-A.

GDC 1 requires quality standards and maintenance of appropriate records. 10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The staff evaluated whether GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed for the information systems important to safety, in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. The staff's evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1) and GDC 1 in Sections 7.1.1.3.3 and 7.1.1.3.6 of this report applies to the information systems important to safety. In RAI 7.5-8, the staff requested the applicant to update references to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," to be consistent with the changes that became effective on September 27, 2007 (e.g., the references in DCD Tier 2, Revision 5, Section 7.5.1.3.1, related to parts of the rule that were deleted). RAI 7.5-8 was being tracked as an open item in the SER with open items. In its response, the applicant corrected citations to the revised rule throughout the DCD. The staff finds the response is acceptable since the applicant cites the appropriate portions of 10 CFR Part 52. Based on the applicant's response, RAI 7.5-8 is resolved. Based on the review of updated DCD information, the staff finds that the DCD Tier 2, Revision 9, properly addresses RG and IEEE standard compliance. The staff finds that requirements of 10 CFR 50.55a(a)(1) and GDC 1 are adequately addressed.

10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995. The staff evaluated whether the information systems important to safety conform to 10 CFR 50.55a(h) and IEEE Std 603. The staff evaluation of IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the information systems important to safety. As described in Sections 7.5.9 and 7.5.10 of the report, for the nonsafety annunciator systems, SPDS, and ERF and ERDS information systems, SRP Section 7.5 indicates that the applicable requirements of 10 CFR 50.55a(h) for IEEE Std 603 are Sections 5.6.3 and 6.3. Section 7.1.1.3.10 of this report provides an evaluation of IEEE Std 603, Sections 5.6.3 and 6.3, that applies to the nonsafety information systems. Based on the review of information documented in DCD Tier 2, Revision 9, Subsections 7.1.6.6.1.7 and 7.1.6.6.1.19, and DCD Tier 1, Revision 9, Section 2.2.15, Item 10, the staff finds that requirements of IEEE Std 603, Sections 5.6.3 and 6.3, are adequately addressed.

The staff evaluated whether 10 CFR 50.34(f)(2)(v), 10 CFR 50.34 (f)(2)(xii), and 10 CFR 50.34 (f)(2)(xiv) are adequately addressed for the information systems important to safety. Section 7.1.1.3.4 of this report evaluates the I&C system design's compliance with 10 CFR 50.34(f)(2)(v), 10 CFR 50.34 (f)(2)(xii), and 10 CFR 50.34 (f)(2)(xiv). This evaluation applies to the information systems important to safety. Accordingly, the staff finds that 10 CFR 50.34(f)(2)(v), 10 CFR 50.34(f)(2)(xii), and 10 CFR 50.34(f)(2)(xiv) are adequately addressed for the information systems important to safety.

The staff evaluated whether 10 CFR 50.34(f)(2), Subparts (v) [I.D.3], (xi) [II.D.3], (xvii) [II.F.1], (xviii) [II.F.2], (xix) [II.F.3], and (xxiv) [II.K.3.23], are adequately addressed for the PAM instrumentation and CMS. Section 7.1.1.3.4 of this report provides an evaluation of these requirements that applies to the PAM instrumentation and CMS. The staff finds that 10 CFR 50.34(f)(2) Subparts (v), (xi), (xvii), (xviii), (xix), and (xxiv) are adequately addressed.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the information systems important to safety. SRP Section 7.2 identifies that GDC 2 and 4 are addressed by the identification of those systems and components for the RPS designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles in the design bases. SRP Section 7.2 also identifies that GDC 2 and 4 are addressed by the review of the qualification program in DCD Tier 2, Sections 3.10 and 3.11. DCD Tier 2, Revision 9, Table 7.1-1 identifies that GDC 2 and 4 apply to the information systems important to safety. DCD Tier 2, Revision 9, Table 3.2-1, identifies that the safety information systems important to safety are designed as seismic Category I systems. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Sections 3.10 and 3.11 of this report, respectively. DCD Tier 1, Revision 9, Table 3.8-1, Items 1 and 3, include the ITAAC for the applicant to verify the EQ of safety electrical and digital I&C equipment. The evaluation of GDC 2 and 4 in Section 7.1.1.3.6 of this report further addresses these topics and applies to the information systems important to safety.

SRP Section 7.5 identifies that the review should verify that the instrumentation provided for monitoring severe accident conditions is designed to operate in the severe accident environment for which it is intended and over the time span for which it is needed.

As discussed in Section 7.5.2.3 of this report, the staff evaluated whether there is reasonable assurance that accident monitoring instrumentation would perform its intended function in severe accident environments. NEDO-33201, Table 8.D.2-1 identifies the required functions and associated monitored variables. NEDO-33201, Section 8D.4.6, provides the equipment capability evaluation of the PAM equipment. In Section 19.2.3.3.8. of this report, the staff provides its evaluation of the severe accident equipment survivability analysis and finds that the analysis provides reasonable assurance that the equipment necessary to achieve a controlled, stable plant condition will function over the time span in which it is needed. Accordingly, based on the applicant's identification of EQ programs consistent with the design bases for the information systems important to safety and their verification in the ITAAC, the staff finds that the requirements of GDC 2 and 4 are adequately addressed.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 13 and 19 and the SRM to SECY-93-087 are adequately addressed for the information systems important to safety. Section 7.1.1.3.6 of this report evaluates the evaluation of GDC 19. SRP Section 7.5 identifies that GDC 13 and 19 are addressed in part by the conformance of the accident monitoring instrumentation to RGs 1.75, 1.97, 1.105, and 1.151. Section 7.5.2.3 of this report evaluates conformance to RG 1.97, which the staff finds acceptable. DCD Tier 2, Revision 9, Table 7.1-1, documents conformance to RGs 1.75, 1.105, and 1.151 for information systems important to safety.

SRP Section 7.5 also identifies that GDC 13 and 19 are addressed in part by verifying that (1) the control room annunciator systems are sufficiently reliable to support normal and emergency plant operations, (2) redundant annunciator systems are provided and the independence of these redundant systems complies with the independence requirements of IEEE Std 603, Section 5.6, (3) alarms provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions comply with the guidance of IEEE Std 603, and (4) the guidance of the SRM to SECY-93-087, Item II.T, is satisfied. The staff evaluated each of these topics in Section 7.5.9.3 of this report and finds them adequately addressed.

SRP Section 7.5 also identifies that GDC 13 and 19 are addressed in part by findings above for individual information systems important to safety. The staff finds that (1) the guidelines of RG 1.97 are adequately addressed for the PAM instrumentation as described in Section 7.5.2.3 of this report, (2) RG 1.47, scope of indications, and independence are adequately addressed for BISI for safety systems as described in Section 7.5.8.3 of this report, and (3) reliability, independence, redundancy and the SRM to SECY-93-087, Item II.T, are adequately addressed as described in Sections 7.5.9.3 and 7.1.1.3.7 of this report.

The findings discussed above and conformance to GDC 13 and 19 are supported by or depend on an identified interrelated process to design and verify the monitoring capability, particularly (1) the inclusion of applicable IEEE Std 603 criteria in the systems' design bases; (2) DCD Tier 1, Revision 9, Section 2.2.15, including the DAC/ITAAC for the applicant to verify conformance to these IEEE Std 603 criteria; (3) DCD Tier 1, Revision 9, Section 3.7, including performance criteria and ITAAC to confirm that the post accident instrumentation is installed consistent with the selected variables; (4) DCD Tier 1, Revision 9, Section 3.3, including the DAC/ITAAC to confirm that the HFE design is implemented based on the process described in DCD Revision 9, Chapter 18; (5) the development of software for the MCR, including the alarms, using the software development activities described in NEDE-33226P and the software QA activities described in NEDE-33245P; and (6) DCD Tier 1, Revision 9, Section 3.2, including the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. These verifications apply to the information systems important to safety and include verification of the controls for manual initiation of functions necessary to support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Based on the above, the staff concludes that requirements of GDC 13 and 19 are adequately addressed for the information systems important to safety.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for the information systems important to safety. SRP Appendix 7.1-A notes that GDC 24 is addressed for protection systems by conformance to IEEE Std 603, Sections 5.1, 5.6, 5.12, 6.3, 6.6, and 8, particularly Sections 5.6 and 6.3. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 24 applies to the information systems important to safety. DCD Tier 2, Revision 9, Sections 7.1.6.6.1.7 and 7.1.6.6.1.19, describe conformance with IEEE Std 603, Sections 5.6 and 6.3. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, including Sections 5.6 and 6.3. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the applicable I&C systems' design is completed in compliance with IEEE Std 603, including Sections 5.6 and 6.3. In addition, SRP Section 7.5 identifies that GDC 24 is addressed in part by the SPDS and the ERF and ERDS information systems, as well as by the nonsafety portions of the accident monitoring instrumentation, BISI, and annunciator systems being appropriately isolated from safety systems. The staff evaluated the independence or isolation of the SPDS and the ERF and ERDS information systems, BISI, and annunciator systems in Sections 7.5.10.3, 7.5.8.3, and 7.5.9.3, and finds them acceptable. Accordingly, based on their conformance to the applicable guidance and IEEE Std 603 and their verification in the DAC/ITAAC, the staff finds that the requirements of GDC 24 are adequately addressed.

The staff evaluated whether the applicant met the requirements of 10 CFR 52.47(b)(1). This regulation requires that the application (for design certification) contain 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. The staff finds the ITAAC specific to the information systems important to safety. which are addressed throughout this section, to be acceptable. Accordingly, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed.

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in SRP Appendix 7.1-D. The staff's evaluation of IEEE Std 7-4.3.2 in Section 7.1.1.3.10 of this report applies to the information systems important to safety, including the safety PAM instrumentation, CMS, and BISI for safety systems.

7.5.12 Conclusion

Based on the above, the staff concludes that the applicant adequately addresses the major design considerations for the information systems important to safety. As discussed in Sections 7.1.1.3.1 through 7.1.1.3.10 of this report and Sections 7.5.2 through 7.5-11 above, the staff concludes that, for the information systems important to safety, the applicant adequately addresses the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2), Subparts (v), (xi), (xvii), (xviii), (xix), (xxiv); 10 CFR 52.47(b)(1); and GDC 1, 2, 4, 13, 19, and 24. The staff also finds that the design of the information systems important to safety meets the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152. The applicant has also identified adequate high-level functions and included sufficient DAC/ITAAC in DCD Tier 1, Revision 9, to verify that the design of the information systems important to safety is completed in compliance with the applicable requirements.

7.6 Interlock Logic

7.6.1 Regulatory Criteria

The objective of the review of the interlock logic is to confirm that design considerations such as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of this logic and commensurate with the importance of the safety functions to be performed.

Acceptance criteria in SRP Section 7.6, Revision 5,for the interlock logic are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(v); 10 CFR 52.47(b)(1); and GDC 1, 2, 4, 13, 19, 24, and 25.

7.6.2 Summary of Technical Information

In DCD Tier 2, Revision 9, Section 7.6, the applicant addressed the high pressure/low pressure (HP/LP) interlock logic. The FAPCS is a low-pressure piping system with the following interfaces to the high pressure RWCU/SDC system.

- Its LPCI line is connected to the RWCU/SDC system Loop B discharge line, which is connected to the RPV via the Feedwater Loop A discharge line.

- Crosstie connections are provided from the FAPCS suppression pool suction to the RPV RWCU line to the regenerative heat exchanger (RHX) (RWCU suction) and from the RWCU return line (discharge line to the RPV) to the FAPCS discharge line to the suppression pool, GDCS pools, and containment spray line.

During reactor power operation, the high pressure in the RWCU/SDC system piping exceeds the design pressure of the low-pressure FAPCS piping. The LPCI line isolation valves consist of parallel pairs of air-operated, testable check valves and motor-operated block valves to protect the FAPCS low-pressure piping from overpressurization during reactor power operation. These valves are normally closed. The testable check valves and the motor-operated valves (MOVs) are nonsafety. Parallel valves are provided for redundancy and fire zone separation. Both sets of parallel valves have identical interlock logic for operation; however, the power supplies for operation of these valves are provided from different sources, the PIP A and PIP B buses, for redundancy and fire zone separation. The logic for operation of the valves is implemented in the PIP A N-DCIS and the PIP B N-DCIS.

The HP/LP interlock logic prevents the isolation valves from opening, and closes them if opened, whenever there is a high pressure signal from the RPV pressure transmitters of the NBS. The high pressure signal also prevents testing of the air-operated testable check valves and closes them if they are open for testing. It also prevents the operation of the LPCI mode of the FAPCS. DCD Tier 2, Revision 9, Section 9.1.3.2, describes the FAPCS modes. An SRV is provided upstream of the LPCI line check valves to protect against overpressurization of the pipe by leakage through the check valves. The relief valve discharge line is monitored to detect any leakage through the check valves. The crosstie from the FAPCS to the RWCU/SDC system is used only following a LOCA. These connections allow the RWCU/SDC system to provide containment cooling after a LOCA to bring the plant to a cold shutdown. Each FAPCS to RWCU/SDC system crosstie connection is isolated with a spectacle flange, a check valve, and an MOV providing a positive isolation. The flange removal and the operation of the crosstie are under administrative control.

The PIP A and PIP B provide the power supplies for nonsafety pressure instruments, logic, and solenoids (for operation of testable check valves). The power supplies for operation of the LPCI line's nonsafety, motor-operated parallel valves are provided from different sources, the PIP A and PIP B buses, for redundancy and fire zone separation. These nonsafety power supplies are backed up by nonsafety batteries and diesel generators.

The high reactor pressure signals from the NBS processed in the N-DCIS are used to determine whether a high pressure condition exists in the RWCU/SDC discharge line to the RPV feedwater inlet line. If a high pressure condition exists, the interlock logic sends a signal to close the MOV. This signal also prevents testing of the check valves and prevents the LPCI mode of operation of the FAPCS.

7.6.3 Staff Evaluation

DCD Tier 2, Revision 9, Section 7.6, describes one interlock to prevent overpressurization of low-pressure systems. DCD Tier 2, Section 7.6, does not include any of the four other interlocks identified in SRP Section 7.6. In RAIs 7.6-1 and 7.6-2, the staff requested that DCD Tier 2, Section 7.6, include all interlock logic important to safety, particularly, the interlock logic to isolate safety systems from nonsafety systems. The RAI responses describe the basis for the one interlock logic and the design feature that has built-in interlock provisions within the Q-DCIS platform design, which the staff finds acceptable. Accordingly, this evaluation only addresses the acceptance criteria for an interlock logic to prevent overpressurization of low-pressure systems. Based on the applicant's responses, RAIs 7.6-1 and 7.6-2 are resolved.

In DCD Tier 2, Revision 9, Section 7.6.1.3, the applicant states that no HP/LP interface involves safety systems. A nonsafety HP/LP interface involves the low-pressure FAPCS LPCI line, which interfaces with a high pressure condition in the RWCU/SDC system piping. The RWCU/SDC system piping interfaces with the feedwater line, which maintains the RCPB.

The FAPCS HP/LP interlock logic prevents the opening of the isolation valves on the LPCI discharge line. The interlock logic prohibits the LPCI line isolation valves from being opened whenever the reactor pressure is greater than the reactor pressure permissive setpoint for the interlock logic, thereby protecting the low-pressure FAPCS piping from overpressurization during reactor power operation. The interlock logic is designed to permit LPCI mode initiation when the reactor pressure is below its reactor pressure permissive setpoint allowing the operator to manually open either isolation valve. The interlock logic operates automatically, and its status is provided to the reactor operator in the MCR and the RSS panels.

The LPCI line provides a path to bring in fire water or suppression pool water for reactor shutdown cooling 72 hours after a DBE, if the normal shutdown cooling system is not available. Therefore, the HP/LP interlock logic is nonsafety and within the scope of regulatory treatment of nonsafety systems (RTNSS). However, the staff found that DCD Chapter 19, Revision 5, Appendix A, regarding RTNSS systems, did not include this item. In RAI 7.6-3, the staff requested the applicant to document this HP/LP interlock logic in DCD Tier 2, Chapter 19, Appendix A. RAI 7.6-3 was being tracked as an open item in the SER with open items. In its response, the applicant stated that the interlock logic exists to protect low pressure piping, but the protection of low pressure piping is not required to meet the RTNSS criteria specified in DCD Tier 2, Chapter 19, Appendix A. The applicant revised DCD Tier 2, Section 7.6.1.3, "Safety Evaluation," to delete references to RTNSS treatment of the "HP/LP Interlock logic." The interlock logic functions are embedded in the DCIS logic such that there is no separate interlock logic. The applicant revised DCD Tier 2, Sections 7.1.3.2.5, 7.1.6.5, and 7.6, to remove references to the phrase, "Interlock System," which it replaced with the phrase, "Interlock Logic." The staff finds the response is acceptable since the applicant clarified the HP/LP interlock logic and incorporated it into DCD Revision 6. Based on the applicant's response, RAI 7.6-3 is resolved.

10 CFR 50.55a(a)(1) requires that structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. SRP Appendix 7.1-A states that the

applicant should commit to conformance with the RGs, codes and standards referenced in SRP Sections 7.1 through 7.9, applicable BTPs, and SRP Appendix 7-A. With regard to conformance with 10 CFR 50.55a(a)(1), DCD Tier 2, Revision 9, Section 7.6.1.3.1 states that the HP/LP interlock logic is nonsafety-related. DCD Tier 2, Section 7.6.1.3 describes the basis for the HP/LP interlock logic being nonsafety-related. The testable check valves provide pressure boundary integrity for the RWCU/SDC system. The motor-operated, normally closed, fail-as-is gate valves provide defense-in-depth protection against any leakage passing through the check valves. A safety relief valve is provided upstream of the testable check valves to protect against over-pressurization of the pipe by leakage through the check valves. Based on the HP/LP interlock logic being applied to nonsafety-related valves, the staff concurs that the HP/LP interlock logic is nonsafety-related.

DCD Tier 2, Revision 9, Section 7.1.4.4, provides the N-DCIS regulatory requirements conformance summary. DCD Tier 2, Revision 9, Section 7.1.6, Table 7.1-1, further describes conformance to applicable portions of the regulations. DCD Tier 2, Table 7.1-1, indicates conformance with 10 CFR 50.55a(a)(1) for network segments PIP A, N-DCIS and PIP B, N-DCIS that implement the HP/LP interlock logic. While DCD Tier 2, Tables 7.1-1, specifies conformances to some but not all of the RGs identified in SRP Appendix 7.1-A, the staff finds that DCD Tier 2, Table 7.1-1, specifies conformance to the RGs applicable to HP/LP interlock logic. The staff finds compliance in the design with 10 CFR 50.55a(a)(1) is adequately addressed for the HP/LP interlock logic.

10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995. With regard to 10 CFR 50.55a(h) compliance with IEEE Std 603, DCD Tier 2, Revision 9, Section 7.6.1.3.1 states that the HP/LP interlock logic is nonsafety-related and thus 10 CFR 50.55a(a)(1) does not apply to the HP/LP interlock logic. As described in the evaluation of 10 CFR 50.55a(a)(1) above, the staff concurs that the HP/LP interlock logic is nonsafety-related and 10 CFR 50.55a(h) is not applicable to the HP/LP interlock logic. Although 10 CFR 50.55a(h) and IEEE Std 603 do not apply to this system, each of the parallel air-operated testable check valves and each of the parallel MOV is powered from either the PIP A or PIP B bus. Similarly, the interlock logic is implemented in the PIP A or PIP B providing separation and isolation, both mechanically and electrically. The staff finds this acceptable.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the HP/LP interlock logic. DCD Tier 2, Revision 9, Table 7.1-1 identifies that GDC 2 and 4 apply to the Q-DCIS and the N-DCIS. DCD Tier 2, Revision 9, Sections 7.6, states that the HP/LP interlock logic is classified as nonsafety equipment and qualified to the environmental conditions existing at the location of the devices. The evaluation of GDC 2 and 4 in Section 7.1.1.3.6 of this report further addresses these topics applicable to the N-DCIS. Accordingly, based on the applicant's identification of EQ programs consistent with the design bases for the interlock logic, the staff finds that the requirements of GDC 2 and 4 are adequately addressed.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 13 and 19 are adequately addressed for the HP/LP interlock logic. Section 7.1.1.3.6 of this report evaluates conformance with GDC 19. DCD Tier 2, Revision 9, Section 7.6, describes the monitoring capability and controls for the HP/LP interlock logic. The interlock logic operates automatically, and its status is provided to the reactor operator in the MCR. The HP/LP interlock logic is designed to permit LPCI mode initiation when the reactor pressure is below its reactor pressure permissive setpoint allowing the operator to manually open either isolation valve. The staff finds these monitoring capabilities and controls acceptable. Based on the review of DCD Tier 2, Revision 9, Section 7.6, documentation, the staff finds that the requirements of GDC 13 and 19 are adequately addressed.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for interlock logic. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 24 applies to all I&C systems, including PIP A and PIP B network segments, which contains the HP/LP interlock logic. The staff evaluation of GDC 24, as described in Section 7.1.1.3.6, Item 13, of this report applies to the HP/LP interlock logic. Accordingly, the staff finds that the requirements of GDC 24 are adequately addressed.

10 CFR 50.34(f)(2)(v) [I.D.3] requires an applicant to provide an automatic indication of the bypassed and operable status of the safety systems. With regard to conformance with 10 CFR 50.34(f)(2)(v)[I.D.3], DCD Tier 2, Section 7.6.1.3.1 states that the HP/LP interlock logic does not have a bypass feature. Because the HP/LP interlock logic cannot be bypassed, the staff finds that 10 CFR 50.34(f)(2)(v) does not apply.

10 CFR 52.47(b)(1) requires that the application for design certification contain 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. With regard to 10 CFR 52.47(b)(1), DCD Tier 2, Section 7.6.1.3.1 states that ITAAC are provided for the I&C systems and equipment in DCD Tier 1. The HP/LP interlock logic is not included in the ITAAC. As described in the evaluation of 10 CFR 50.55a(a)(1) above, the staff concurs that the HP/LP interlock logic is nonsafety-related. In addition, the HP/LP interlock logic is not required by any GDC. Based on the above, the staff finds that ITAAC are not needed for the HP/LP interlock logic and the HP/LP interlock logic meets the requirements of 10 CFR 52.47(b)(1)

GDC 25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. With regard to GDC 25, DCD Tier 2, Revision 9, Section 7.6.1.3.2 states that because the HP/LP interlock logic does not involve reactivity control, GDC 25 does not apply. The staff concurs that the HP/LP interlock logic does not involve reactivity control.

7.6.4 Conclusion

As discussed in Section 7.6.3 above, the staff concludes that the HP/LP interlock logic meets the relevant requirements of 10 CFR 50.55a(a)(1); GDC 1, 2, 4, 13, 19, and 24; and 10 CFR

52.47(b)(1). Furthermore, the staff concludes that 10 CFR 50.55a(h), 10 CFR 50.34(f)(2)(v) and GDC 25 do not apply to the HP/LP interlock logic.

7.7 Control Systems

DCD Tier 2, Revision 9, Section 7.7, describes I&C systems for normal plant operations that do not perform plant safety functions. However, these systems do control plant processes that have an impact on plant safety and control. This includes the main reactivity control of the nuclear reactor core with the positioning of the control rods, control of feedwater to the RPV, feedwater temperature, and regulation of reactor steam flow and pressure. Both of these systems can force the actuation of the safety functions and prevent the need for the safety functions to actuate either through normal operation or through inadvertent operation, or various AOOs.

While not directly essential to the safe shutdown and maintenance of the nuclear reactor and plant in a safe condition, these systems must not prevent the safety function from operating when required. Further, the ability of these systems to meet the acceptance criteria also depends on quality software and human factors development which are outside the scope of the evaluation in this section. The control systems described in this section include the following:

- NBS(N) – nonsafety subsystems
- RC&IS
- FWCS
- PAS
- SB&PC system
- NMS(N) – nonsafety subsystems
- CIS

The nonsafety instruments and controls of the RC&IS, FWCS, PAS, SB&PC system, NMS(N), and NBS(N) are part of a group of systems that are collectively grouped with the N-DCIS. The controls for the CIS are not part of the N-DCIS, but have direct controls in the MCR. The relationship of these systems with other nonsafety systems and with safety systems is indicated in a simplified network functional diagram of the DCIS in DCD Tier 2, Revision 9, Figure 7.1-1. DCD Tier 2, Revision 9, Figure 7.1-3, provides a distributed power-sensor diversity diagram. DCD Tier 2, Revision 9, Figure 7.1-4, provides a hardware/software (architecture) diversity diagram. The N-DCIS is segmented into five parts that can work independently of one another if failures occur as outlined in DCD Tier 2, Revision 9, Sections 7.1.4.8 and 7.1.5.2. One of these five segments is the BOP segment, which involves Section 7.7 control systems with a single channel of triple-redundant controllers that execute the functions of the SB&PC system, the PAS, the FWCS, and the FWTCS. The GENE network segment executes the functions of the RC&IS with dual-redundant controllers.

7.7.0 Evaluation of Common Aspects of Control Systems

7.7.0.1 *Regulatory Criteria*

The objective of the review of DCD Tier 1, Revision 9, Section 2.2, and DCD Tier 2, Revision 9, Section 7.7, is to confirm that (1) the control systems comply with the regulations by conforming to the applicable acceptance criteria and guidelines, (2) the controlled variables can be maintained within prescribed operating ranges, and (3) the effects of operation or failure of these systems are bounded by the accident analyses in DCD Tier 2, Revision 9, Chapter 15.

Acceptance criteria in SRP Section 7.7, Revision 5, for control systems are based on meeting the relevant requirements described in SRP Section 7.7 as 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); 10 CFR 50.34(f)(2)(xxii); 10 CFR 52.47(b)(1); and GDC 1, 10, 13, 15, 19, 24, 28, 29, and 44. The acceptance criteria are also based on conforming to the guidelines of the SRM to SECY-93-087. For nonsafety control systems evaluated in this section, 10 CFR 50.34(f)(2)(xxii) is identified as an acceptance criteria only for B&W plants and therefore does not apply to the ESBWR.

7.7.0.2 *Common Control System Acceptance Criteria*

The staff reviewed the control systems below in accordance with SRP Section 7.7. The staff also used acceptance criteria in SRP Table 7-1, SRP Appendix 7.1-A, and SRP Appendix 7.1-C, as directed by SRP Section 7.7. The staff used the regulatory criteria listed in Section 7.7.0.1 of this report as the basis for the review of the control system discussed in DCD Tier 2, Revision 9, Section 7.7.

These systems are classified as nonsafety, since they are not depended on for safe shutdown of the reactor or for maintaining it in a safe condition. However, in accordance with BTP HICB-19, these systems are considered the first echelon of defense in avoiding situations in which the safety reactor protection systems must respond. Therefore, this safety evaluation will note the significant features of these control systems that enhance the overall concept of nuclear power plant safety, support D3, and aid in the avoidance of spurious actuation.

7.7.0.3 *Evaluation of Control Systems Conformance with Common Acceptance Criteria*

The control systems of DCD Tier 2, Revision 9, Section 7.7, are associated with the N-DCIS and share many basic design and safety attributes. This section will evaluate the acceptance criteria that are common to all or nearly all of these control systems. Exceptions, acceptance criteria that apply to a specific control system, or special features that enhance safety will be discussed under the staff evaluation for the applicable section of specific control systems. In accordance with SRP Section 7.7, the following major design considerations, which are common to all of the control systems in Section 7.7, unless otherwise indicated, should be emphasized in the control systems review and the staff evaluation:

(1) Design Basis

SRP Section 7.7 states for design bases that the review should confirm that the control systems include the necessary features for manual and automatic control of process variables within prescribed normal operating limits. This acceptance criterion is system dependent and is reviewed under the staff evaluation for the specific control system within Sections 7.7.1 through 7.7.7 of this report.

(2) Safety Classification

SRP Section 7.7 states for safety classification that the review should confirm that the plant accident analysis in Chapter 15 does not rely on the operability of any control system function to assure safety. The staff reviewed DCD Tier 2, Revision 9, Chapter 15, and verified that Chapter 15 does not rely on DCD Tier 2, Revision 9, Section 7.7, control systems to assure safety. In particular, none of the AOOs and accidents identified in DCD Tier 2, Revision 9, Chapter 15, Tables 15.1-5 and 15.1-6, rely on the control systems to scram the reactor and maintain it in a safe condition. As discussed in the specific control system evaluation below, these control systems often are able to help mitigate an event to avoid the reactor trip system from actuating. In addition, DCD Tier 2, Revision 9, Sections 7.1.6.6.1.7 and 7.1.6.6.1.19, state that the Q-DCIS protection systems are separate and independent from the nonsafety control systems, in accordance with GDC 24, and that any failure of nonsafety systems does not affect safety systems or prevent them from performing their safety functions. Section 7.1.1.3.6 of this report evaluates the conformance of the Q-DCIS to GDC-24.

In RAIs 7.7-7 and 7.7-8, the staff requested the applicant to clarify the safety classification of the NBS I&C. DCD Tier 2, Revision 5, Section 7.7, described both the safety and nonsafety portions of the NBS I&C, but the design of the nonsafety portion is not clear. RAIs 7.7-7 and 7.7-8 were being tracked as open items in the SER with open items. In its responses, the applicant revised DCD Tier 2, Section 7.7, to discuss only the nonsafety portions of the NBS I&C and moved the discussion of the safety portions to the applicable sections in DCD Tier 2, Chapter 7. The staff finds the responses are acceptable since the applicant clarified the design of the nonsafety portion of the NBS I&C. Based on the applicant's response, RAIs 7.7-7 and 7.7-8 are resolved.

Based on the above, the staff finds that the safety classification of nonsafety systems is correctly designated for the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

(3) Effects of Control System Operation on Accidents

SRP Section 7.7 states that the review of the effects of control system operation on accidents should confirm that the safety analysis considers the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOOs. This acceptance criterion is system dependent and is reviewed under the staff evaluation for the specific control system within Sections 7.7.1 through 7.7.7 of this report.

(4) Effects of Control System Failures

SRP Section 7.7 states that the review of the effects of control system failures should confirm that the failure of any control system component or any auxiliary supporting system for control systems does not cause plant conditions more severe than those described in the analysis of AOOs in DCD Tier 2, Revision 9, Chapter 15. This acceptance criterion is system dependent and is reviewed under the staff evaluation for the specific control system within Sections 7.7.1 through 7.7.7 of this report.

(5) Effects of Control System Failures Caused by Accidents

SRP Section 7.7 states that the review should confirm that the consequential effects of AOOs and accidents do not lead to control system failures that would result in consequences more severe than those described in the analysis in DCD Tier 2, Revision 9, Chapter 15. This acceptance criterion is system dependent and is reviewed under the staff evaluation for the specific control system within Sections 7.7.1 through 7.7.7 of this report.

(6) Environmental Controls in Control Systems

SRP Section 7.7 states that the review should confirm that I&C systems include environmental controls as necessary to protect equipment from environmental extremes.

Table 7.7-1. Conformance to Environmental Controls in Control Systems.

RG/GDC	NBS(N)	RC&IS	FWCS	PAS	SB&PC	NMS(N)	CIS
1.89	1	1	1	1	1	1	1
1.97	C	C	C	C	C	C	C
1.100	2	2	2	2	2	2	2
1.151	C		3		4		C
1.180	C,1	C,1	C,1	1	1	C,1	C,1
1.204	C						
1.209	1	1	1	1	1	1	1
GDC 2	C	C	C	C	C	C	C,5
GDC 4	C	C	C	C	C	C	C,5

Notes:
C = Conforms with the RG or GDC indicated for specified control system,
1 = See DCD Tier 2, Revision 9, Table 3.11-1,
2 = See DCD Tier 2, Revision 9, Sections 3.9 and 3.10,
3 = Receives signals from sensors on RPV instrument lines in the NBS(N),
4 = Not applicable. Receives signals from sensors in NBS(N) and other systems,
5 = CIS instrument lines penetrating containment comply with GDC 2 and 4

The staff reviewed DCD Tier 2, Revision 9, Sections 3.9, 3.10, 3.11, 7.7, and Table 3.11-1, to identify environmental controls for the control systems discussed in DCD Tier 2, Revision 9, Section 7.7. DCD Tier 2, Revision 9, Section 7.7.1.3, states that the NBS(N) instruments are designed to operate under normal and peak operating conditions of system pressure and at ambient pressures and temperatures. Based on conformance to the RGs and GDC, as noted above in Table 7.7-1, the staff finds conformance to environmental controls adequately addressed for the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

(7) Use of Digital Systems

SRP Section 7.7 states that control system software should be developed using a structured process similar to that applied to safety system software. In Section 7.1.2.3.7 of this report, the staff finds that NEDE-33226P and NEDE-33245P provide a structured process for respectively developing and assuring the quality of control system software, which is appropriately tailored for safety systems and the important nonsafety systems of DCD Tier 2, Revision 9, Section 7.7, and is therefore acceptable. Accordingly, the staff finds that the use of digital systems for the control systems evaluated throughout Section 7.7 of this report, including the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS, is adequately addressed.

(8) Independence

See discussion of IEEE Std 603-1991 under Item (2) of Section 7.7.0.4 of this report.

(9) Diversity and Defense-in-Depth

SRP Section 7.7 states that control system elements credited in the D3 analysis should be reviewed using the criteria for the DPS described in SRP Section 7.8. In RAI 7.7-11, the staff requested that the applicant clarify which nonsafety systems perform diverse functions. In response, the applicant revised the DCD to state in DCD Tier 2, Section 7.1.6.3 that the digital

I&C systems are designed for high reliability to minimize the potential for CCFs by applying principles of D3. Further, DCD Tier 2, Section 7.1.5.3, states that the nonsafety portions of the systems that conform to the SRM to SECY-93-087, Item II.Q, and BTP HICB-19 are discussed in DCD Tier 2, Section 7.8., The staff evaluated this topic in Section 7.8 of this report. The applicant addressed D3 in NEDO-33251, which the staff evaluated in Section 7.1.3.3 of this report. The staff confirmed the changes were incorporated into DCD Revision 8. The staff finds that the response is acceptable since the applicant clarified which nonsafety systems perform diverse functions. Based on the applicant's response, RAI 7.7-11 is resolved. Features that contribute to D3 are discussed in the sections for specific control systems.

(10) Potential for Inadvertent Actuation

SRP Section 7.7 states that the control systems design should limit the potential for inadvertent actuation and challenges to safety systems. The staff reviewed the control systems discussed in DCD Tier 2, Revision 9, Section 7.7, to identify design measures that limit the potential for inadvertent actuation. Any examples of such design features are discussed for the specific control systems within Sections 7.7.1 through 7.7.7 of this report.

(11) Control of Access

SRP Section 7.7 states that physical and electronic access to digital computer-based control system software and data should be controlled to prevent changes by unauthorized personnel. SRP Section 7.7 further states that the control should address access via network connections and maintenance equipment. In RAI 7.7-13, the staff requested that the applicant clarify the access controls for control systems. In response, the applicant revised DCD Tier 2, Section 7.1.5.1.2, to specifically provide key-locked control equipment cabinet doors including position switches, and electronic protection of control systems, including password protection. The staff confirmed that the applicant incorporated the changes into DCD Revision 8. The staff finds that the response is acceptable since the applicant clarified the access controls for control systems. Based on the applicant's response, RAI 7.7-13 is resolved. This conclusion applies to the control systems evaluated throughout Section 7.7 of this report, which include the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

7.7.0.4 *Evaluation of Control Systems Common Compliance with Regulations*

The control systems associated with the N-DCIS share many basic design and quality attributes that contribute to a safer plant. This section will evaluate the compliance with regulations that are common to all or nearly all of these control systems. Exceptions, conformance with regulations that only apply to a specific control system, or special features that enhance plant safety are reviewed within the staff evaluation for the specific control system under Sections 7.7.1 through 7.7.7 of this report.

In accordance with SRP Section 7.7, the following regulations must be met by all control systems in DCD Tier 2, Revision 9, Section 7.7, unless otherwise indicated:

(1) Compliance with 10 CFR 50.55a(a)(1)

The regulation at 10 CFR 50.55a(a)(1) states, "Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed." SRP Appendix 7.1-A states that the applicant should commit to conformance with the RGs, codes, and standards referenced in SRP Sections 7.1 through 7.9, applicable BTPs, and SRP Appendix 7-A. SRP Appendix 7.1-A further states that the design should conform to all RGs and industry standards committed to by the applicant and that 10 CFR 50.55a(a)(1) applies to all safety and nonsafety I&C systems including the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

DCD Tier 2, Revision 9, Section 7.1.4.4, provides the N-DCIS regulatory requirements conformance summary, and DCD Tier 2, Revision 9, Section 7.1.6, Table 7.1-1, further describes conformance to applicable portions of the regulations. DCD Tier 2, Revision 9, Table 7.1-1, indicates conformance with 10 CFR 50.55a(a)(1) for network segments for GENE and BOP that support and interface with the control systems of DCD Tier 2, Revision 9, Section 7.7. The applicant's safety evaluation for each of the control systems of DCD Tier 2, Section 7.7, indicates compliance with 10 CFR 50.55a(a)(1) by conformance and use of the applicable standards. While DCD Tier 2, Tables 7.1-1, specifies conformance to some, but not all of the RGs identified in SRP Appendix 7.1-A, the staff finds that DCD Tier 2, Table 7.1-1, specifies conformance to the RGs applicable to these nonsafety systems. The staff finds compliance in the design with 10 CFR 50.55a(a)(1) is adequately addressed for the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

(2) Compliance with 10 CFR 50.55a(h) Requiring Conformance to IEEE Std 603-1991

10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995. SRP Appendix 7.1-A states that, for nonsafety systems isolated from safety systems the applicable requirement of 10 CFR 50.55a(h) for IEEE Std 603-1991 is Section 5.6.3, "Independence Between Safety Systems and Other Systems." SRP Table 7.1 indicates that the requirement to provide separation between protection and control functions (Sections 5.6.3 and 6.3.1) applies to all I&C systems. DCD Tier 2, Revision 9, Table 7.1-1, indicates conformance with 10 CFR 50.55a(h) for network segments for GENE and BOP that support and interface with the control systems of DCD Tier 2, Revision 9, Section 7.7. The applicant's safety evaluation for each of the control systems of DCD Tier 2, Section 7.7, indicates compliance with 10 CFR 50.55a(a)(1). Section 7.1.1.3.10 of this report provides an evaluation of IEEE Std 603, Sections 5.6 and 6.3, which applies to the control systems. In Section 7.1.1.3.10 of this report, the staff finds that Sections 5.6 and 6.3 of IEEE Std 603 are adequately addressed based on their inclusion in the safety systems' design basis and their confirmation in the DAC/ITAAC. Accordingly, the staff finds that the requirements of 10 CFR 50.55a(h) are adequately addressed for the control systems discussed in DCD Tier 2, Revision 9, Section 7.7, including the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

(3) Compliance with 10 CFR 52.47(b)(1)

This regulation requires that the application for design certification contain proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the ITAAC are performed and the acceptance criteria are met, a plant that references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations. Compliance with this regulation and the identification of the DCD Tier 1 ITAAC is reviewed under the staff evaluation for the specific control system within Sections 7.7.1 through 7.7.7 of this report.

(4) Compliance with GDC 1, 10, 13, 15, 19, 24, 28, 29, and 44

Table 7.7-2. ESBWR GDC Conformance List for Control Systems of DCD Tier 2, Section 7.7.

GDC	NBS(N)	RC&IS	FWCS	PAS	SB&PC	NMS(N)	CIS
1 R	C	C	C	C	C	C	C
2	C	C	C	C	C	C	C
4	C	C	C	C	C	C	C
10 R							
12		C				C	
13 R	C	C	C	C	C	C	C
15 R							
19 R	C	C	C	C	C	C	C
20	C						
21	C						
22	C						
23	C						
24 R	C	C	C	C	C	C	C
25						C	
26						C	
27						C	
28 R		C				C	
29 R		C				C	
41							C
42							C
43							C
44 R							
Note: C = Conforms per DCD; R = Required by SRP Section 7.7 (if applicable)							

GDC 1 requires quality standards and maintenance of appropriate records. SRP Appendix 7.1-A states that the applicant should commit to conformance with the applicable RGs, codes, and standards referenced in SRP Sections 7.1 through 7.9, the BTPs, and SRP Appendix 7-A. DCD Tier 2, Revision 9, Table 7.1-1, indicates conformance with GDC 1 for network segments for GENE and BOP that support and interface with the control systems of DCD Tier 2, Revision 9, Section 7.7. In RAI 7.7-14, the staff requested that the applicant clarify conformance to GDC 1 for control systems. RAI 7.7-14 was being tracked as an open item in the SER with open items. In its response, the applicant indicated that it would revise the safety evaluation for each of the control systems discussed in DCD Tier 2, Section 7.7, to indicate compliance with GDC 1 by conformance with and use of the applicable industry standards. The staff confirmed that the applicant incorporated these changes into DCD Revision 8. Table 7.7-2 summarizes the conformance of control systems in DCD Tier 2, Section 7.7, to applicable GDC. The staff finds that the response is acceptable since the applicant clarified the conformance to GDC 1 for control systems. Based on the applicant's response, RAI 7.7-14 is resolved. While DCD Tier 2, Revision 9, Table 7.1-1, specifies conformance to some, but not all of the RGs identified in SRP Appendix 7.1-A, the staff finds that DCD Tier 2, Table 7.1-1, specifies conformance to RGs applicable to these nonsafety systems. The staff finds the design's compliance with GDC 1 is adequately addressed for the NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 10 will be addressed in the evaluation for each control system in sections 7.7.1 through 7.7.7 of this report.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 13 will be addressed in the evaluation for each control system in sections 7.7.1 through 7.7.7 of this report.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. GDC 15 will be addressed in the evaluation for each control system in sections 7.7.1 through 7.7.7 of this report.

GDC 19 requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. GDC 19 will be addressed in the evaluation for each control system in sections 7.7.1 through 7.7.7 of this report.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. SRP Table 7.1 indicates that GDC 24 applies to all control systems. The staff evaluated whether GDC 24 is adequately addressed for the N-DCIS control systems. DCD Tier 2, Revision 9, Sections 7.1.6.6.1.7 and 7.1.6.6.1.19, describe conformance with IEEE Std 603, Sections 5.6 and 6.3. These sections state that the Q-DCIS protection systems are separate and independent from the nonsafety control systems, in accordance with GDC 24, and that any failure of nonsafety systems does not affect safety systems or prevent them from performing their safety functions. Section 7.1.1.3.10 of this report evaluates conformance with IEEE Std 603, Sections 5.6 and 6.3, and finds that Sections 5.6 and 6.3 are adequately addressed based on their inclusion in the Q-DCIS design basis and their verification in the DAC/ITAAC. Accordingly, based on the appropriate isolation of the control systems from the safety systems and the inclusion of IEEE Std 603, Sections 5.6 and 6.3, in the design basis for the Q-DCIS and their verification in the DAC/ITAAC, the staff finds that the requirements of GDC 24 are adequately addressed for the N-DCIS control systems (NBS(N), RC&IS, FWCS, PAS, SB&PC, NMS(N), and CIS).

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RPV internals to impair significantly the capability to cool the core. GDC 28 will be addressed in the evaluation for each control system in sections 7.7.1 through 7.7.7 of this report.

GDC 29 requires that protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. GDC 29 will be addressed in the evaluation for each control system in sections 7.7.1 through 7.7.7 of this report.

GDC 44 requires a system be provided to transfer heat from SSCs important to safety, to an ultimate heat sink. As described in Section 7.1.1.3.6 of this report, the staff finds that GDC 44 is not directly applicable to the control systems of DCD Tier 2, Revision 9, Section 7.7.

7.7.1 The Nuclear Boiler System Instrumentation and Control - (Nonsafety Subsystems)

7.7.1.1 *Summary of Technical Information*

The NBS(N) provides the monitoring and control input for important nuclear reactor related variables during normal plant operating modes and during the plant response to accidents. The NBS sensors used for safety system actuation and control functions are addressed in other sections within this chapter. This section describes the NBS(N) used for indication, actuation, and control of nonsafety systems.

The NBS(N) provides RPV water level and dome pressure measurements over the ranges, and to the accuracies necessary for adequate operator monitoring of RPV water level during normal, transient, and accident conditions. The NBS(N) provides indication of the following parameters in support of normal plant operations and power generation, as well as during AOOs and events as needed:

- Reactor coolant and RPV temperatures
- RPV water level
  - Shutdown range
  - Narrow range
  - Wide range
  - Fuel zone range
- RPV pressure
- SRV discharge line temperature
- Main steam flow rate

The NBS(N) design provides for periodic calibration and testing of the NBS(N) instrumentation during plant operation. Nonsafety instruments are powered from the nonsafety instrument power supply buses.

The NBS(N) provides the following measurement and monitoring, as discussed in DCD Tier 2, Revision 9, Section 7.7.1.2.2:

(1) Reactor Coolant and Reactor Pressure System Vessel Temperature Monitoring

The reactor coolant temperatures are measured at the mid-vessel inlet to the RWCU/SDC system and at the bottom head drain. Coolant temperature can also be determined in the steam-filled parts of the RPV and steam-water mixture by measuring the reactor pressure. In the saturated system, reactor pressure connotes saturation temperature. Core inlet temperature can normally be measured by the redundant core inlet temperature sensors located in each LPRM assembly below the core plate elevation.

The RPV outside surface temperature is measured at the head flange and at the bottom head locations.

(2) Reactor Pressure Vessel Water Level

The RPV level instruments are differential pressure devices calibrated for the specific vessel pressure and liquid temperature conditions. The method of water level measurement is the condensing chamber reference leg type, which uses differential pressure devices as its primary elements. The reactor water level measurement is temperature compensated through the thermocouples installed on the sensing line. Reactor water level instrumentation is used to (1) provide signal input to the FWCS and (2) provide signal inputs for the DPS functions as discussed in DCD Tier 2, Revision 9, Section 7.8.1. DCD Figure 7.7-1 provides a diagram indicating the range and RPV tap points. DCD Tier 2, Revision 9, Section 7.7.1.2.2 describes four ranges of water level instrumentation.

- The shutdown range water level instrumentation is used to monitor the RPV water level during shutdown conditions when the head is removed and the reactor system may be flooded for refueling or maintenance.

- The narrow-range water level instrumentation is used to monitor the RPV water level for use with the FWCS for normal operation and abnormal events.

- The wide-range water level instrumentation is used for safety and nonsafety applications and for the DPS, and is provided for the range of normal, transient, and accident conditions. The wide-range water level measurement has its own separate sensors and indicators.

- The fuel zone range water level instrumentation is provided for PAM in which the water level may be substantially below the normal range. The maximum point limit uses the RPV taps near the top of the steam outlet nozzle. The fuel zone range water level measurement has its own separate sensors and indicators.

(3) Reactor Pressure Vessel Pressure

Pressure transmitters detect RPV pressure from the instrument lines used for measuring RPV water level to provide indication and status in the MCR.

(4) Safety Relief Valve Leak Detection

Thermocouples are located in the discharge pipes of 10 SRVs. The temperature signals are recorded, and temperatures indicative of a leaking SRV are alarmed in the MCR.

(5) Main Steam Flow Rate

Differential pressure transmitters are used to determine the steam flow rate. Pressure taps from the throat of the RPV steam outlet nozzles, in conjunction with the RPV dome pressure taps, measure differential pressure. The square root of the differential pressure is proportional to the main steam flow rate and is used for feedwater control.

7.7.1.2 *Staff Evaluation*

7.7.1.2.1 Evaluation of NBS(N) Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations in accordance with SRP Section 7.7, and discusses the attributes that are common to the control systems of DCD Tier 2, Revision 9, Section 7.7. This section discusses and evaluates only those major design considerations and information that are unique to the NBS(N).

(1) Design Basis

DCD Tier 2, Revision 9, Section 7.7.1.1.2, does not identify any manual or automatic control functions for the NBS(N). The NBS(N) provides indication of the parameters needed in support of normal plant operations. DCD Tier 2, Revision 9, Section 7.7.1.5, lists these parameters and states that they are displayed in the MCR. In addition, the RPV pressure is indicated at four local instrument racks in the reactor building. Based on the above, the staff finds that the design bases are adequately addressed for the nonsafety NBS(N).

(2) Effects of Control System Operation on Accidents

DCD Tier 2, Revision 9, Section 7.7.1.1.2, identifies that the NBS(N) does not have any manual or automatic control functions. The NBS(N) is only a monitoring system. The NBS(N) monitors process parameters that are used by the FWCS and SB&PC system and are indicated in the MCR. The effect of NBS(N) action on transients is to continue to provide process information to the FWCS and SB&PC system, as well as to the MCR, which is acceptable to the staff. The effect of NBS(N) inaction on transients is bounded by the NBS(N) failures evaluated below and the staff finds it to be acceptable. Accordingly, the staff finds that the effects of NBS(N) system operation on accidents are adequately addressed.

(3) Effects of Control System Failures

The NBS(N) monitors process parameters used by the FWCS and the SB&PC system so NBS(N) failures (including digital, sense, and transmission failures) are bounded by FWCS failures and SB&PC failures, the dominant failure from the FWCS. DCD Tier 2, Revision 9, Sections 15.2.4.2, 15.3.1, and 15.3.2, analyze three bounding FWCS failures: runout of one feedwater pump, loss of feedwater heater with failures of Select Control Rod Run-in (SCRRI) and Select Rod Insertion (SRI), and feedwater controller failure with maximum flow demand. These failures bound the failures of the FWCS and thus the NBS(N). Further, the applicant stated in DCD Tier 2, Revision 9, Section 7.7.1.3, that, if a line break should occur in a nonsafety portion of a sensing line, the excess flow check valve closes to stop the flow of reactor coolant, and, if there is a single failure of the excess flow check valve, a restriction orifice limits the flow of coolant to within acceptable bounds. Accordingly, the staff finds that the NBS(N) system failures do not cause plant conditions more severe than those described in the analysis of AOOs in Chapter 15.

In RAI 7.7-12, the staff requested that the applicant provide analyses that evaluate the effects of control systems failures. RAI 7.7-12 was being tracked as an open item in the SER with open items. In response, the applicant revised the DCD Tier 2, Section 7.7 in the discussion of specific control systems to reference DCD Tier 2, Chapter 15 analyses of specific events that evaluate the effects of control systems failures. The applicant stated that since the outputs from the NBS(N) transmitters are used by other systems such as the FWCS, the references to DCD

Tier 2, Chapter 15, analyses in DCD Tier 2, Section 7.7.3.3, for failure modes of the FWCS also apply to failure modes of the NBS(N). Thus, DCD Tier 2, Section 7.7.1.3 does not include references to DCD Tier 2, Chapter 15, analyses for failures of the NBS(N). The applicant also stated that although credit is not taken for nonsafety systems in the safety functions, failure of these nonsafety systems does not prevent safe shutdown of the reactor. The applicant revised the DCD Tier 2, Revision 9, Section 7.7 to reference DCD Tier 2, Revision 9, Chapter 15 analyses of specific events that evaluate the effects of control systems failures. As discussed above, the expected and abnormal transients and accident events analyzed in DCD Tier 2 Revision 9, Sections 15.2.4.2, 15.3.1, and 15.3.2, bound the failure modes associated with the FWCS digital controls and thus failure in the NBS(N). The staff finds the response is acceptable since the applicant identified the DCD Tier 2, Revision 9, Chapter 15 analyses that bound the failures of the NBS(N). Based on the applicant’s response, RAI 7.7-12 regarding the NBS(N) is resolved.

(4) Effects of Control System Failures Caused by Accidents

A potential effect of an accident on the NBS(N) would be to damage the instrumentation sensing lines in such a manner as to significantly disrupt the NBS(N) output signals to the FWCS or SB&PC system, thus affecting the FWCS or SB&PC system control functions. The use of multiple divisions and independent sense lines significantly reduces the probability of such effects. In the previous section, the staff finds that NBS(N) system failures do not cause plant conditions more severe than those described in the analysis of AOOs in DCD Tier 2, Revision 9, Chapter 15. AOOs and accidents do not lead to more severe NBS(N) failures because the applicant assumed maximum failures of the NBS(N) in the Chapter 15 analyses. Based on the above, the staff finds that the effects of NBS(N) system failures caused by accidents are adequately addressed.

(5) Potential for Inadvertent Actuation

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.1, to identify design measures that limit the potential for inadvertent actuation. Examples of such design features are discussed below.

The NBS(N) is a measurement system for providing information to other systems, such as the FWCS and SB&PC system, and does not involve actual control of equipment or processes. Thus, the potential for inadvertent activation is reduced compared to a system like the FWCS, which provides controls. Inadvertent activation could occur based on incorrect information from the NBS(N), but the probability of this occurring is significantly reduced by the use of multiple independent sensors and transmitters through multiple divisions and the use of self-test, diagnostics, and parameter value comparison.

The NBS(N) is designed with redundancy so that failure of any single instrument does not result in the loss of level and pressure indication. The RPV water level, main steam flow rate, feedwater flow, and temperature measurement use multiple signals provided by the NBS(N), which are displayed and alarmed in the MCR.

Redundant sensors located in each LPRM assembly below the core plate elevation measure the core inlet temperatures. This provides both redundant measurements, as well as a radial distribution of the core inlet temperatures. As presented earlier, RPV water level is measured by four physically separate level (differential pressure) transmitters mounted on separate divisional local racks in the safety envelope within the reactor building. Each transmitter is on a separate pair of instrument lines and is associated with a separate RPS electrical division.

Each division has its own set of RPV sensing line nozzle connections. Further, there are four ranges of RPV water level instruments. Based on the above, the staff finds that the NBS(N) design limits the potential for inadvertent actuation.

7.7.1.2.2 Evaluation of NBS(N) Compliance with Regulations

Section 7.7.0.4 of this report lists and discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for the control systems identified in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the NBS(N).

(1) Compliance with 10 CFR 52.47(b)(1)

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the NBS(N). DCD Tier 1, Revision 9, Section 2.2, does not have a specific table of contents entry for the NBS(N). Instead, since the NBS(N) is supplying parameter readings to other systems and is associated with the N-DCIS network, the NBS(N) is considered to be tested and accepted when the ITAAC of the systems requiring the NBS(N) parameter data are completed. The ITAAC for the systems using NBS(N) data are found in DCD Tier 1, Revision 9, Sections 2.2.3, 2.2.5, 2.2.9, 2.2.11, 2.2.14, and 2.2.15. Based on the review of DCD Tier 2, Revision 9, Section 7.7.1; DCD Tier 1, Revision 9, Sections 2.2.3, 2.2.5, 2.2.9, 2.2.11, 2.2.14, and 2.2.15; and the ITAAC of the systems using NBS(N) data, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the NBS(N).

(2) Compliance with GDC 10, 13, 15, 19, 28, and 29

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.1, to verify that the applicable GDC specified in SRP Section 7.7 are adequately addressed for the NBS. DCD Tier 2, Revision 9, Section 7.7.1.3.2, states that the NBS(N) complies with GDC 13 and 19.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the NBS(N). SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. The applicant does not credit compliance with GDC 10 for the NBS(N). DCD Tier 2, Revision 9, Section 7.7.1.1.2, identifies that the NBS(N) does not have any manual or automatic control functions. The NBS(N) provides essential measurements of reactor pressure, RPV water level, reactor coolant, RPV temperatures, main steam flow rate, and SRV discharge temperatures. Accordingly, the staff finds that GDC 10 does not apply to the NBS(N).

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the NBS(N). DCD Tier 2, Revision 9, Section 7.7.1.1.2, identifies that the NBS(N) does not have any manual or automatic control functions. DCD Tier 2, Revision 9, Section 7.7.1.3.2, indicates conformance to GDC 13. The NBS(N) provides essential measurements of reactor pressure, RPV water level, reactor coolant and RPV temperatures, main steam flow, and SRV discharge temperatures. Accordingly, based on information reviewed in DCD Tier 2, Revision 9, Section 7.7.1, and their verification in the ITAAC of systems that receive inputs from the NBS(N), the staff finds that the requirements of GDC 13 are adequately addressed for the NBS(N).

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the NBS(N). SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate the I&C system contributions to design margin for reactor coolant systems. The applicant does not credit compliance with GDC 15 for the NBS. DCD Tier 2, Revision 9, Section 7.7.1.1.2, does not identify any manual or automatic control functions for the NBS(N). The NBS(N) monitors process parameters that are used by the FWCS and SB&PC system and are indicated in the MCR. Since the NBS(N) does not control any parameter that may affect reactor margins, the staff finds that GDC 15 does not apply to the NBS(N).

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the NBS(N). In Section 7.1.1.3.6 of this report, the staff evaluated whether GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should evaluate if I&C are available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.1.3.2, specifies that the NBS(N) conforms to GDC 19. As described in DCD Tier 2, Revision 9, Section 7.7.1.5, the process parameters monitored by the NBS(N) are displayed in the MCR. DCD Tier 2, Revision 9, Section 7.7.1.1.2, identifies that the NBS(N) does not have any manual or automatic control functions. Based on the above, the staff finds that the NBS(N) provides information to permit actions to be taken to operate the plant safely during normal operation and accidents. Accordingly, the staff finds that the requirements of GDC 19 are adequately addressed for the NBS(N).

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 28 and 29 are adequately addressed for the NBS(N). The NBS(N) monitors process parameters that are used by the FWCS and the SB&PC system and are indicated in the MCR. Accordingly, the NBS(N) is not a reactivity control system and the staff finds that GDC 28 and 29 do not apply to the NBS(N).

Based on the above, the staff finds that the NBS(N) adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 above for a nonsafety system and that there is reasonable assurance that this system will be able to accomplish its designed function in a reliable manner when built and tested according to DCD Tier 2, Revision 9, and the DCD Tier 1, Revision 9, ITAAC.

7.7.1.3 *Conclusion*

Based on the above, the staff concludes that (1) there is reasonable assurance that the NBS(N) conforms to the applicable requirements, which include GDC 1, 10, 13, 19, and 24, 10 CFR 52.47(b)(1); 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h); (2) adequate high level functional requirements are identified; and (3) sufficient ITAAC of systems using the output from the

NBS(N) are included in DCD Tier 1, Revision 9, to verify that the design is completed in compliance with the applicable requirements.

7.7.2 Rod Control and Information System

7.7.2.1 *Summary of Technical Information*

As described in DCD Tier 2, Revision 9, Section 7.7.2, the main objective of the RC&IS is to control the Fine Motion Control Rod Drive (FMCRD) motors of the CRD system to permit changes in core reactivity so that reactor power level and power distribution can be controlled. The RC&IS obtains status and control rod position information from the CRD FMCRD instrumentation. It sends purge water valve control signals to, and obtains status signals from, the HCUs of the CRD system. The RC&IS sends and receives status and control signals to and from other plant systems and RC&IS modules. The RC&IS also monitors and assists in excess feedwater temperature change protection using the ATLM subsystem.

The RC&IS consists of multiple types of cabinets, or panels, that contain special electronic/electrical equipment modules for performing the RC&IS logic in the reactor building and control building. It also includes a dedicated operator interface (DOI) on the main control panel in the MCR. The RC&IS DOI provides summary and status information to the plant operator with respect to control rod positions, FMCRD, RC&IS status, and HCU status. The RC&IS also provides controls for performing normal rod movement functions, bypassing major RC&IS subsystems, performing CRD surveillance tests (except the FMCRD holding brake testing performed during a refueling outage), and resetting RC&IS trips and most abnormal status conditions. A few abnormal status conditions require reset actions at local control panel equipment. DCD Tier 2, Revision 9, Section 7.7.2.1.2, lists the functions performed by the RC&IS.

Nine types of electronic/electrical cabinets/panels perform the logic functions of the RC&IS:

- Rod action control subsystem (RACS) cabinet
- Remote communication cabinet (RCC)
- Induction motor controller cabinet (IMCC)
- Rod brake controller cabinet (RBCC)
- Emergency rod insertion control panel (ERICP)
- Emergency rod insertion panel (ERIP)
- Scram time recording panel (STRP)
- Scram time recording and analysis panel (STRAP)
- RAPI auxiliary panels

The RC&IS scope includes the following equipment:

- All the electrical/electronic equipment contained in the RACS cabinet, the RCCs, the remote communication cabinets, the IMCCs, the RBCCs, the STRPs, the STRAP, the ERIPs, and the ERICP (note: RAPI auxiliary panels are designated as part of the N-DCIS)

- The RC&IS multiplexing network equipment

- The cross-channel communication links between equipment located in the RACS cabinets

- The dedicated RC&IS DOI and the communication links from the RACS cabinets to DOI interface

7.7.2.2 *Staff Evaluation*

7.7.2.2.1 Evaluation of Rod Control and Information System Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations noted in SRP Section 7.7 and discusses the attributes that are common to the control systems of DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those major design considerations and information unique to the RC&IS.

(1) Design Basis

With regard to the design basis, the applicant stated in DCD Tier 2, Revision 9, Section 7.7.2.1.1, that the RC&IS has no functional safety design basis and is designed so that the functional capabilities of safety systems are not inhibited. DCD Tier 2, Revision 9, Section 7.7.2.1.2, describes the functions performed by the RC&IS, including changes to core reactivity through controls that position the control rods in manual, semiautomatic, and automatic modes of operation. Examples of these functions include the (1) scram-follow function, (2) automatic enforcement of rod movement blocks and enforcement of adherence to predetermined rod patterns, (3) manual and automatic insertion of all control rods by an alternate and diverse method, (4) insertion of selected control rods upon SCRRI/SRI command signals from the DPS, (5) enforcement of fuel operating thermal limits, (6) calculation of a reference feedwater temperature used in feedwater temperature rate of change control, and (7) surveillance test support. Further, the RC&IS provides control rod position and FMCRD and RC&IS status summary information through dedicated operator interface in the MCR. Based on the above and the verification of these controls and functions through the DCD Tier 1, Revision 9, Section 2.2.3, ITAAC, the staff finds that the design bases are adequately addressed for the RC&IS.

(2) Effects of Control System Operation on Accidents

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.2 and Chapter 15, to evaluate the effects of the RC&IS operation on accidents. DCD Tier 2, Section 7.7.2.1.2, describes many operational functions performed by the RC&IS, which attempt to prevent or mitigate transients, but a failure of the RC&IS does not prevent safe shutdown of the reactor. Examples include the automatic enforcement of rod movement blocks to prevent potentially undesirable rod movements and a possible scram. These blocks do not affect the hydraulic scram insertion function, the scram-follow function, the ARI function, or the SCRRI function. Further, the RC&IS operation provides automatic, electric motor run-in of all operable control rods, following detection of activation of the hydraulic insertion of the control rods by a reactor scram. The RC&IS inserts select control rods upon SCRRI/SRI command signals from the DPS. If the feedwater temperature decreases by more than 16.7 degrees Celsius (C) (30 degrees Fahrenheit [F]) from the reference feedwater temperature calculated by the ATLM, the RC&IS (via the ATLM) provides a feedwater temperature control valve one-way block, rod withdrawal block, and SCRRI/SRI initiation. The staff reviewed DCD Tier 2, Revision 9, Chapter 15, Tables 15.1-5, 15.1-6, and 15.1-7, which summarize the creditable bounding AOOs, and other events, transients, and accidents. DCD Tier 2, Table 15.1-5, credits the SCRRI and SRI with assisting in mitigating the following transients: Loss of Feedwater Heating (LOFWH), Generator

Load Reject with Bypass, Turbine Trip with Bypass, Generator Load Rejection with a Single Failure in the Bypass System, and Turbine Trip with a Single Failure in the Bypass System. The Rod Block function is credited with assisting in mitigating the transients that involve control rod withdrawal error. Since these RC&IS protective features are automatic, the applicant concludes that failure of the RC&IS to perform is a failure of one or more components or input information as described below. Failure of the RC&IS does not cause plant conditions more severe than those described in the analysis of AOOs in DCD Tier 2, Revision 9, Chapter 15. Based on the previous discussions, as well as those that follow, the staff finds that the safety analysis considers the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOOs. The staff finds this acceptable.

(3) Effects of Control System Failures

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.2 and Chapter 15, regarding the effects of RC&IS failures. The staff verified that the failure of the RC&IS does not cause plant conditions more severe than those described in the analysis of AOOs in DCD Tier 2, Revision 9, Chapter 15. The failure of the RC&IS is bounded by considering the worst case of failures involving (1) failure of I&C components or failure of input, (2) an operator error in the manual positioning of the control rods, and (3) failure of an RC&IS indication that causes the operator to make an error in manually positioning the control rods.

The consequences of component or system type failures of the RC&IS are limited since the RC&IS directly controls movement of each control rod or rod gang. A failure that results in inadvertent movement of a control rod affects only one control rod or rod gang. The malfunctioning in positioning of any single control rod or rod gang does not impair the effectiveness of a reactor scram. Therefore, no single failure in the RC&IS prevents a reactor scram. Repair, adjustment, or maintenance of the RC&IS enforces all rod blocks until the rod block condition is cleared. The applicant stated that the circuitry described for the RC&IS is independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the RC&IS circuitry from affecting the scram circuitry.

Another potential result of the failure of a component or system of the RC&IS could be the failure to send the “signal to initiate” to prompt the N-DCIS to initiate the SCRRI and SRI functions. Such failures of the RC&IS are bounded and discussed in DCD Tier 2, Revision 9, Chapter 15. Again, unless other systems act as the MRBM or the operators detect and correctly identify the source of a problem with the RC&IS and take mitigating action in time, the RPS acts to protect the reactor from failure of the RC&IS. With regards to manual positioning errors, DCD Tier 2, Revision 9, Section 7.7.2.2.7.4, lists 16 types of conditions in which either one channel or both channels of the RC&IS logic receives a signal that will cause it to issue a rod block, preventing further rod movement until the situation is corrected. During startup and below the low-power setpoint, the RWM enforces preplanned, analyzed, and preloaded control rod sequences called reference rod pull sequence (RRPS). The ATLM protects against exceeding fuel parameter limits on minimum critical power ratio (MCPR) and maximum linear heat generation rate (MLHGR). The MRBM subsystem protects against regional high neutron flux. The safety SRNM and APRM of the NBS have rod blocks before reactor trip setpoints are reached. Other protective rod blocks come from I&C hardware or power failures; unacceptable parameter situations such as CRD charging low water pressure; and operational situations, such as refueling platform over the core. The RC&IS enforces all rod blocks until the rod block condition is cleared. The bypass capabilities of the RC&IS permit clearing certain rod block conditions caused by failures or problems that exist in only one channel of the logic. With the proper functioning of the RPS, a failure of the RC&IS is controlled.

DCD Tier 2, Revision 9, Chapter 15, identifies occurrences and events related to RC&IS and control rod movement, as listed in DCD Tier 2, Revision 9, Chapter 15, and Tables 15.0-2, 15.1-3, 15.1-5, 15.1-6, and 15.1-7. The staff finds that these occurrences and events are consistent with the staff's identified failures of the RC&IS and, therefore, finds that the failure of the RC&IS does not cause plant conditions more severe than those described in DCD Tier 2, Revision 9, Chapter 15. In RAI 7.7-12, the staff requested that the applicant provide analyses that evaluate the effects of control systems failures. RAI 7.7-12 was being tracked as an open item in the SER with open items. In response, the applicant revised the DCD Tier 2, Section 7.7 to reference DCD Tier 2, Chapter 15 analyses of specific events that evaluate the effects of control systems failures. As discussed above, the expected and abnormal transients and accident events analyzed in DCD Tier 2, Sections 15.2.3.1, 15.2.3.2, 15.3.8 and 15.3.9 bound the effects of RC&IS failures. The staff finds that the response is acceptable since the applicant identified the DCD Tier 2, Revision 9, Chapter 15 analyses that bound the failures of the RC&IS. Based on the applicant's response, RAI 7.7-12 regarding the RC&IS is resolved.

(4) Effects of Control System Failures Caused by Accidents

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.2 and Chapter 15, regarding the effects of the RC&IS system failures caused by accidents. The RC&IS consists of multiple types of cabinets, or panels, that contain special electronic/electrical equipment modules for performing the RC&IS logic in the reactor building and control building. Accidents could potentially damage the sensors and transmitters providing input to the RC&IS as in the NBS(N) system. Fire or earthquake potentially could cause one or more functions to fail and damage the electronic equipment causing an RC&IS function to fail. The potential for such effects is significantly reduced through D3 in the RC&IS sensors, the RC&IS EQ, redundant modules in different cabinets, and a fire protection design. Regardless, in the event of failure of the RC&IS, the applicant states in DCD Tier 2, Revision 9, Section 7.7.2.2.1, that a failure or malfunction of the RC&IS has no effect on the hydraulic scram function of the CRD. The circuitry for normal insertion and withdrawal of control rods in the RC&IS is independent of the RPS circuitry controlling the scram valves. This separation of the RPS scram and the RC&IS normal rod control functions prevents a failure in the RC&IS circuitry from affecting the scram circuitry. RC&IS failures are bounded by the following DCD Tier 2, Revision 9, Chapter 15 transients from Table 15.1-7: (1) Control Rod Withdrawal Error During Power Operation with ATLM Failures and (2) Control Rod Withdrawal Error During Startup With Failure of Control Rod Block. Based on the above, the staff finds that the consequential effects of AOOs and accidents do not lead to RC&IS failures that would result in consequences more severe than those described in DCD Tier 2, Revision 9, Chapter 15. Based on the above, the staff finds that the effects of RC&IS failures caused by accidents are adequately addressed.

(5) Potential for Inadvertent Actuation

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.2, to identify design measures that limit the potential for inadvertent actuation. Examples of such design features are described below.

The RC&IS enforces all rod blocks until the rod block condition is cleared. The bypass capabilities of the RC&IS permit clearing certain rod block conditions caused by failures or problems that exist in only one channel of the logic.

The RC&IS uses a dual redundant architecture of two independent channels for normal monitoring of control rod positions and executing normal control rod movement commands. Under normal conditions, each channel receives separate input signals, and both channels perform the same functions. The outputs of the two channels are continuously compared. For normal functions of enforcing and monitoring control rod positions and emergency rod insertion, the outputs of the two channels must agree. Any sustained disagreement between the two channels results in a rod block. However, when the conditions for generating a rod block signal in a single channel are satisfied, that channel alone (independent of the other channel) issues a rod block signal.

The design allows the RC&IS to continue to operate, when practical, in the presence of component hardware failures. This is achieved because the operator is able to reconfigure the operation of the RC&IS through bypass capabilities while the failures are being repaired. No single power source or single power supply failure results in the loss of the RC&IS functions.

Based on the above, the staff finds that the RC&IS design limits the potential for inadvertent actuation. The staff notes that the applicant stated in DCD Tier 2, Revision 9, Section 7.7.2.2.7.5, that the expected reliability is based upon the expected frequency of an inadvertent movement of more than one control rod due to failure. The expected frequency is less than or equal to one inadvertent movement in 100 reactor operating years.

7.7.2.2.2 Evaluation of Rod Control and Information System Compliance with Regulations

Section 7.7.0.4 of this report lists and discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for the control systems identified in DCD Tier 2, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the RC&IS.

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the RC&IS. The staff reviewed DCD Tier 2, Revision 9, Section 7.7.2, and DCD Tier 1, Revision 9, Section 2.2.1, in accordance with SRP Sections 7.7 and 14.3.5, to verify that 10 CFR 52.47(b)(1) is adequately addressed for the RC&IS. DCD Tier 1, Section 2.2.1, documents the RC&IS ITAAC requirements. While the RC&IS has no DAC, the RC&IS methods and functions of controlling the control rod positioning, the general equipment and modes of control, and the actuation initiators of protective actions are specified. Accordingly, based on information reviewed in DCD Tier 1, Revision 9, Section 2.2.1; DCD Tier 2, Revision 9, Chapter 7 and Section 7.7.2; information discussed herein; and identified RC&IS I&C and their verification in the ITAAC, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed.

To determine compliance with GDC 10, 13, 15, 19, 28, and 29, the staff reviewed DCD Tier 2, Revision 9, Section 7.7.3, to verify that the applicable GDC specified in SRP Section 7.7 are adequately addressed for the RC&IS. DCD Tier 2, Revision 9, Section 7.7.1.3.2, states that the RC&IS complies with GDC 13 and 19.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the RC&IS. SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. DCD Tier 2, Revision 9, Chapter 15, and Tables 15.1-5 and 15.1-6, identify RC&IS actuations and other actions that reduce the need for the actuation of safety systems to mitigate AOOs. DCD Tier 2, Revision 9, Section 7.7.2.1.2, includes corresponding actions in the power generation (nonsafety) design bases of the RC&IS to maintain the reactor core, reactor coolant system, and the reactivity limits within appropriate margins and to mitigate AOOs. Examples of such design features are described below.

The control rod block functions provide appropriate margin to protect the reactor core by stopping control rod movement before the RPS is required to initiate a scram. The RC&IS mitigates AOOs by inserting selected control rods to counteract the positive reactivity effects of a loss of feedwater heating event or provide needed power reduction after a load reject event or turbine trip. The ATLM, a subsystem of the RC&IS, helps enforce core thermal limits and feedwater temperature rate of decrease. The RWM forces predefined and approved low rod-worth rod patterns during low-power operations. DCD Tier 1, Revision 9, Section 2.2.1, includes the ITAAC for the applicant to verify that the as-built RC&IS implements these actions. Accordingly, based on identified RC&IS actions and the verification in the ITAAC, the staff finds that the requirements of GDC 10 are adequately addressed for the RC&IS.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the RC&IS. DCD Tier 2, Revision 9, Section 7.7.2, and DCD Tier 1, Revision 9, Section 2.2.1, identify I&C provided to monitor, control, and maintain the variables over their anticipated ranges for normal operation, for AOOs, and for accident conditions, as appropriate to help assure adequate safety. For the RC&IS, the I&C includes instrumentation for parameters and controls that affect reactivity. The staff reviewed the plant transient response to normal load changes and AOOs, such as control rod withdrawal, control rod drop accident, and control withdrawal error during refueling. The staff concludes that the RC&IS is capable of maintaining system variables within prescribed operating ranges and implementing contingency actions if such variables reach limiting conditions. Examples of I&C are status data on RC&IS components, FMCRD status, and control rod position data, which are collected and provided to other systems. Other examples are the ATLM subsystem, which enforces fuel operating limits and feedwater temperature rate of decrease, the RWM subsystem, which ensures that patterns of control rods are consistent with specific control rod pattern restrictions, such as control rod worth, rod block function, and summary information provided by the RC&IS to the DOI in the control room. DCD Tier 2, Revision 9, Section 7.7.2.3.2, indicates conformance to GDC 13. Accordingly, based on identified RC&IS I&C and their verification in the ITAAC, the staff finds that the requirements of GDC 13 are adequately addressed for the RC&IS.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the RC&IS. SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate I&C system contributions to design margin for reactor coolant systems. DCD Tier 2, Revision 9, Section 7.7.2.3.3, excludes the RC&IS from conformance with GDC 15. The main purpose of the RC&IS is to provide reactivity control through positioning of the control rods; to control the FMCRD motors of the CRD in positioning the control rods; to provide the rod block function; and to provide status information on the CRD, HCU, and control rod positions. Therefore, the RC&IS does not directly contribute to the design margins for the RCPB. Accordingly, GDC 15 does not apply to the RC&IS.

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the RC&IS. In Section 7.1.1.3.6 the staff evaluated that GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should evaluate if there exists I&C available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.2.3.2, specifies that the RC&IS conforms to GDC 19. The features for manual and automatic control described in DCD Tier 2, Revision 9, Section 7.7.2, and DCD Tier 1, Revision 9, Section 2.2.1, identify the RC&IS I&C that facilitate the capability to maintain plant variables within prescribed operating limits and over their anticipated ranges for normal operation, for AOOs, and for accident conditions as appropriate to help assure adequate safety. Examples include the automatic, semiautomatic, and manual control rod controls described in DCD Tier 2, Revision 9, Section 7.7.2.2.7. DCD Tier 2, Revision 9, Section 7.4.2.5, specifies that the parameters displayed and/or controlled from Division 1 and Division 2 in the MCR also are displayed and/or controlled from either of the RSS panels. Section 7.4.3 of this report evaluates the RSS. Accordingly, based on identified RC&IS I&C and the verification in the ITAAC, the staff finds that the requirements of GDC 19 are adequately addressed for the RC&IS.

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. The staff evaluated whether GDC 28 is adequately addressed for the RC&IS. SRP Appendix 7.1-A states that GDC 28 imposes functional requirements on I&C interlock and control systems to the extent they are provided to limit reactivity increases to prevent or limit the effect of reactivity accidents. DCD Tier 2, Revision 9, Section 7.7.2.3.2, specifies that the RC&IS conforms to GDC 28. DCD Tier 2, Revision 9, Section 7.7.2.1.2, summarizes the functions of the design performed by the RC&IS, including those that specifically address GDC 28 and that reduce the need for the actuation of safety systems to mitigate AOOs. These include the rod block functions (including RWM and ATLM), rod scram testing function, SCRRI and SRI, and the scram follow-function. In particular, DCD Tier 2, Revision 9, Section 7.7.2.2.7.4, identifies the control rod block functions performed by the RC&IS. If the feedwater temperature decreases by more than 16.7 degrees C (30 degrees F) from the reference feedwater temperature calculated by ATLM, the RC&IS (via ATLM) provides a feedwater temperature control valve one-way block, rod withdrawal block, and SCRRI/SRI initiation. Also, the RC&IS has special bypass features that allow the operator to perform restricted scram time surveillance testing at any power level, but with no effect on the protective functions for ARI, SCRRI, SCRAM-follow condition, or critical rod block functions, such as provided by the MRBM, APRM, or SRNM. DCD Tier 1, Revision 9, Section 2.2.1, includes the ITAAC for the applicant to verify that the as-built RC&IS implements these actions. Accordingly, based on identified RC&IS actions and their verification in the ITAAC, the staff finds that the requirements of GDC 28 are adequately addressed for the RC&IS.

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 has been adequately addressed. SRP Appendix 7.1-A identifies that GDC 29 is addressed by conformance, as applicable, to GDC 20-25 and GDC 28. Because the RC&IS is a reactivity control system and not a protection system, GDC 20-23 and 25, which are requirements for protection systems, do not apply to the RC&IS. Accordingly, GDC 29 is addressed by conformance as applicable to GDC 24 and 28. DCD Tier 2, Revision 9, Table 7.1-1 and Section 7.7.2.3.2, specifies that the RC&IS conforms to GDC 29. Section 7.7.0.4 of this report evaluates conformance of N-DCIS control systems, including the RC&IS, to GDC 24. Conformance of the RC&IS to GDC 28 is evaluated above. Because the requirements of GDC 24 and 28 are adequately addressed for the RC&IS, the staff finds that the requirements of GDC 29 are adequately addressed for the RC&IS.

Therefore, the staff finds that the RC&IS, adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 of this report for a nonsafety system and that there is reasonable assurance that this system will be able to accomplish its designed function in a reliable manner, when built and tested according to DCD Tier 2, Revision 9, and the DCD Tier 1, Revision 9, ITAAC.

7.7.2.3 *Conclusion*

Based on the above, the staff concludes that (1) there is reasonable assurance that the RC&IS conforms to the applicable requirements, which include GDC 1, 10, 13, 19, 24, 28, and 29, and 10 CFR 52.47(b)(1), 10 CFR 50.55a(a)(1) and 10 CFR 50.55a(h); (2) adequate high level functional requirements are identified, and (3) sufficient ITAAC are included in DCD Tier 1, Revision 9, Section 2.2.1, to verify that the design is completed in compliance with the applicable requirements.

7.7.3 Feedwater Control System

7.7.3.1 *Summary of Technical Information*

The FWCS, as described in DCD Tier 2, Revision 9, Section 7.7.3, regulates the flow of feedwater into the RPV to maintain predetermined water level limits during transients and normal plant operating modes and to control the feedwater temperature. The FWCS is a power generation (control) system that maintains proper RPV water level in the operating range from high (Level 8) to low (Level 3). During normal operation, feedwater flow is delivered to the RPV through three reactor feed pumps (RFPs) which operate in parallel. Each RFP is driven by an adjustable-speed induction motor that is controlled by an adjustable speed drive (ASD) circuit. In normal operation, the fourth RFP is in standby mode and starts automatically if any operating feedwater pump trips while at power.

During normal operation, the FWCS sends three speed-demand signals, each of which reflects a voted FWCS processor output, to each feed pump ASD. The ASD performs a midvalue vote and uses it to control the speed/frequency of the feed pump motor. The midvalue vote is also returned to the FWCS as an analog input and compared with the speed demands sent by the FWCS.

As explained in DCD Tier 2, Revision 9, Section 7.7.3.2.2, the operator can select any one of the following three system operation modes for RPV level control from the main control console: (1) single element control, (2) three-element control, or (3) manual feed pump control. Feedwater temperature control is accomplished by manipulating the heating steam flow to the seventh stage feedwater heaters or directing a portion of the feedwater flow around the high pressure feedwater heaters, as shown in DCD Tier 2, Revision 9, Figure 7.7-7. The two functions are performed by two sets of triple redundant controllers located in separate cabinets, as illustrated in DCD Tier 2, Revision 9, Figure 7.7-3.

Feedwater temperature control has two operational modes according to DCD Tier 2, Revision 9, Section 7.7.3.2.3: (1) Manual, where the feedwater temperature setpoint is controlled by the operator, and (2) Automatic, where the feedwater temperature setpoint is controlled by the PAS. The redundantly measured feedwater temperatures are compared with the temperature setpoint and the error signal is used by a proportional, integral, derivative (PID) controller. The output signals are used to generate the position demands for both the feedwater heater bypass valves and the seventh stage feedwater heater steam heating valves.

The FWCS consists of the following elements, as explained in DCD Tier 2, Revision 9, Section 7.7.3.5.2:

- The FTDC, which contains the software and processors for execution of the control algorithms;

- Feedwater flow signals that provide for the measurement of the total flow rate of feedwater into the vessel;

- Steam flow signals that provide for the measurement of the total flow rate of steam leaving the vessel;

- Feed pump discharge flow signals that provide for the measurement of the discharge flow rate of each feed pump;

- The low flow control valve (LFCV) differential pressure transmitters that provide for the measurement of the pressure drop across the LFCV, for LFCV gain control;

- The LFCV flow transmitters, which provide for the measurement of the flow rate through the LFCV, for both LFCV control and low thermal power calculations; and

- Feedwater temperature signals, which provide the measurement of the feedwater temperature at the point prior to the feedwater piping penetration to the reactor building.

The FWCS includes the following measurement systems described in DCD Tier 2, Revision 9, Sections 7.7.3.5.3 through 7.7.3.5.5:

- Reactor vessel water level measurement
- Steam flow measurement
- Feedwater flow rate measurement

7.7.3.2 *Staff Evaluation*

7.7.3.2.1 Evaluation of Feedwater Control System Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations, in accordance with SRP Section 7.7, and includes the attributes that are common to the control systems discussed in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those major design considerations and information that are unique to the FWCS.

(1) Design Basis

With regard to the design basis, the FWCS is not a safety system and is not required for safe shutdown of the plant. However, the FWCS can have a significant effect on reactivity and thus reactor power. The applicant stated in DCD Tier 2, Revision 9, Section 7.7.3.1.2, that the FWCS is designed so that the functional capabilities of safety systems are not inhibited. The design basis of the FWCS has I&C to regulate the flow of feedwater into the RPV, automatically or manually, to maintain the RPV water level such that predetermined limits on water level are met during transients and normal plant operating modes. Further, the FWCS controls feedwater temperature to allow reactor power control without moving control rods. Based on information reviewed in DCD Tier 1, Revision 9; DCD Tier 2, Revision 9, Section 7.7.3 and Chapter 15; and evaluations that follow, the staff concurs with the classification and finds that the plant's accident analysis does not require operability of the FWCS for safe shutdown of the nuclear power plant. The staff finds that the FWCS includes the necessary features for manual and automatic control of process variables within prescribed operating limits. For example, there is a maximum allowable feedwater temperature setpoint change which cannot be exceeded. Feedwater temperature cannot be decreased when the reactor thermal power exceeds 100 percent. The system does not accept a temperature setpoint outside of the area allowed by the reactor power versus feedwater temperature map described in DCD Tier 2, Revision 9, Section 4.4.4.3. Accordingly, based on the above and the verification of these controls and functions through the DCD Tier 1, Revision 9, Section 2.2.3, ITAAC, the staff finds that the design bases are adequately addressed for the FWCS.

(2) Effects of Control System Operation on Accidents:

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.3 and Chapter 15, for the effects of control system operation on accidents. The FWCS regulates the feedwater flow and feedwater temperature. The FWCS has a primary function in maintaining the proper RPV water level during transients, unless the system is unavailable because of isolation of the feedwater lines, loss of power to the feedwater pumps, or major circuit failure. For example, the FWCS controls RPV water level during a postulated inadvertent isolation condenser initiation or a runout of one feedwater pump. The FWCS initiates a runback of feedwater pump feedwater demand to zero and closes the LFCV and RWCU/SDC Overboard Control Valve (OBCV) upon receiving an ATWS trip signal from the ATWS/SLC Logic, as described further in DCD Tier 2, Revision 9, Section 7.8.1.1. Although no credit is taken for the function in the safety analysis, the feedwater temperature control function also mitigates inadvertent feedwater temperature changes in either direction by manipulating its control valves to maintain the setpoint temperature. In the event of a failure involving the physical control valves, the FWCS will attempt to maintain the feedwater temperature at the setpoint. The temperature difference between feedwater lines A and B is monitored and alarmed if it exceeds the allowable value. The condition in which the FWCS cannot maintain the feedwater temperature is bounded by the transient LOFWH. Based on the information previously discussed, as well as presented below, and as described in DCD Tier 2, Revision 9, Section 7.7.3 and Tables 15.1-5 and 15.1-6, the staff finds that the safety analysis presented in DCD Tier 2, Revision 9, Chapter 15, considers the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOOs.

(3) Effects of Control System Failures:

The staff reviewed DCD Tier 2, Revision 9, Chapter 15 and Section 7.7.3, regarding the effects of FWCS failures. The failure of the FWCS may affect feedwater flow or feedwater temperature. A total failure of either the feedwater flow control subsystem or temperature control subsystem is unlikely because of the use of triple redundant digital controllers, redundant UPS power supplies, and independent input/output signals to each of the three channels (process controllers) of each controller. The probability of a combined feedwater temperature change and feedwater flow/reactor water level change caused by controller failure is significantly reduced or even precluded by implementing the two control schemes in physically different cabinets and logic processors. No single failure or operator error of the FWTCS results in a decrease of more than 55.6 degrees C (100 degrees F) in the final feedwater temperature.

The worst case of a feedwater pump ASD controller failure in the FWCS system would cause a run-out of one feedwater pump to its maximum flow rate. This event would be detected by high feedwater flow, and the FWCS would respond by reducing the demand to the other pumps, automatically compensating for the excessive flow rate from the failed pump.

While unlikely, the total failure of the FWCS and its control of feedwater flow is bounded by (1) failure in a manner that significantly reduces feedwater flow when feedwater is still needed, (2) failure in a manner to oversupply feedwater, and (3) failure in a manner that is still near demand, but no longer responding.

In the first case, RPV water level will decrease, and unless the operators detect and identify the problem correctly and take contingency action, the RPS will actuate. Specifically, if the water falls to Level 3, then the RPS shuts down the reactor. If the water continues to drop and reaches Level 2, the high pressure makeup function of the CRD system initiates (the CRD system is fully independent of other plant delivery or injection systems). The ICS, as part of the ECCS, typically starts operating automatically upon low reactor water level (Level 2), with time delay, and low reactor water level (Level 1), with no delay. The GDCS and ADS, as part of the ECCS, also operate automatically upon low reactor water Level 1. DCD Tier 2, Revision 9, Chapter 15, Table 15.1-6, identifies that if the event involves loss of non-emergency ac power to station auxiliaries, an anticipatory scram occurs.

In the second case, RPV water level will increase unless the operators detect and correctly identify the problem and take contingency action. If the RPV water level rises to Level 8, then the RPS shuts down the reactor. Additionally, the main turbine trips, the feedwater pump ASD flow demand is reduced to zero, and the LD&IS closes the safety feedwater isolation valves. If the RPV water level rises to Level 9, DPS trips the feedwater pumps, and the LD&IS interrupts the ASD controller power supply. In DCD Tier 2, Revision 9, Chapter 15, this event is bounded by the “Feedwater Controller Failure - Maximum Flow Demand,” transient.

In the third case, uncontrolled feedwater flow will eventually drive the RPV water level either too high or too low, but perhaps over such a long period that the possible results may be the same as those in case 1 or 2. However, it is likely that the longer time period significantly increases the probability that the operators will detect and correctly identify the problem and take contingency action long before the high pressure makeup function of the CRD system initiates or protective functions actuate.

The ESBWR uses feedwater temperature as an additional power controlling parameter, as described in DCD Tier 2, Revision 9, Section 7.7.3.2.3. Loss of feedwater heating increases core inlet subcooling and results in greater core power because of increased moderation. While unlikely, the total failure of the FWCS and its control of feedwater temperature is still limited by protective actions. If the reactor thermal power versus feedwater temperature combination is outside of the area allowed by the reactor power versus feedwater temperature map, the RC&IS initiates a control rod withdrawal block and feedwater temperature control valve one-way block. If the reactor thermal power versus feedwater temperature combination further departs from the area allowed by the reactor power versus feedwater temperature map (high reactor thermal power, high feedwater temperature, or low feedwater temperature), the RPS can shut down the reactor.

Both the ATLM and the DPS independently monitor a loss of feedwater heating. If a significant decrease in feedwater temperature is detected by the ATLM or by the DPS, either will mitigate the event by initiating SCRRI and SRI functions. Although no credit is taken for the function in a safety analysis, the FWTCS also mitigates inadvertent feedwater temperature changes in either direction by manipulating its control valves to maintain the setpoint temperature. Further, the temperature difference between feedwater lines A and B is monitored and alarmed if it exceeds the allowable value.

DCD Tier 2, Revision 9, Chapter 15, identifies feedwater occurrences and events related to feedwater flow control and feedwater temperature control in Tables 15.0-2, 15.1-3, 15.1-5, 15.1-6, and 15.1-7. Accordingly, the staff finds that these occurrences and events are consistent with the identified flow control and temperature control failures of the FWCS; therefore the staff finds that the failure of the FWCS does not cause plant conditions more severe than those described in DCD Tier 2, Revision 9, Chapter 15. In RAI 7.7-12, the staff requested that the applicant provide analyses that evaluate the effects of control systems failures. RAI 7.7-12 was being tracked as an open item in the SER with open items. In response, the applicant revised the DCD Tier 2, Section 7.7 to reference DCD Tier 2, Chapter 15 analyses of specific events that evaluate the effects of control systems failures. As discussed above, the expected and abnormal transients and accident events analyzed in DCD Tier 2, Sections 15.2.4.2, 15.3.1 and 15.3.2 bound the effects of FWCS failures. The staff finds the response is acceptable since the applicant identified the DCD Tier 2, Chapter 15 analyses that bound the failures of the FWCS. Based on the applicant's response, RAI 7.7-12 regarding the FWCS is resolved. While the focus in these cases is on feedwater flow, RPV water level, and feedwater temperature, detection of a FWCS control problem and response could come from other parameters including power, pressure, steam flow, and the turbine/generator parameters as well as the FWCS self-diagnostics.

(4) Effects of Control System Failures Caused by Accidents:

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.3 and Chapter 15, regarding the effects of FWCS failures caused by accidents. The FWCS consists of multiple types of cabinets, or panels, that contain special electronic/electrical equipment modules for performing the FWCS logic in the reactor building and control building. Similar to the NBS(N) system, accidents could potentially damage the sensors and transmitters providing input to the FWCS. Fire or earthquake potentially could cause one or more functions to fail and damage the electronic equipment, causing an FWCS function to fail. The potential for such effects is significantly reduced through (1) D3 in the NBS(N) sensors, (2) FWCS environmental equipment qualification, (3) triple redundant controller modules in different cabinets, (4) the fire protection design, (5) a combined feedwater temperature change, and (6) the fact that feedwater flow/reactor water level change caused by a controller failure is precluded by implementing the two control schemes in physically different cabinets and logic processors. Based on the above, as well as the evaluation and review of accident consequences and effects of AOOs on the FWCS, the staff finds that the consequential effects of AOOs and accidents do not lead to FWCS failures that would result in consequences more severe than those described in DCD Tier 2, Revision 9, Chapter 15. Based on the above, the staff finds that the effects of FWCS failures caused by accidents are adequately addressed.

(5) Diversity and Defense-in-Depth:

Section 7.7.0.3 of this report discusses the D3 design consideration. Further, the staff notes that, in support of 10 CFR 50.62 and to mitigate ATWS events, the FWCS initiates a runback of the feedwater pump's feedwater demand to zero and closes the LFCV and RWCU/SDC overboard flow-control valve when it receives an ATWS trip signal from the ATWS/SLC logic. With less feedwater for mixing, core inlet subcooling decreases. This introduces negative reactivity from reduced moderator density and voiding to reduce power. Section 7.8.3 of this report evaluates conformance with 10 CFR 50.62. DCD Tier 2, Revision 9, Section 7.7.3.3.3, states that the portions of the FWCS that provide interface support for the DPS conform to the criteria of the SRM to SECY-93-087, Item II.Q.

(6) Potential for Inadvertent Actuation:

The staff reviewed DCD Tier 2, Revision 9, Chapter 7.7.3 to identify design measures that limit the potential for inadvertent actuation. Examples of such design features are described below.

The FWCS is designed with redundancy so that failure of any single instrument does not interfere with the system operation. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. Each FTDC channel executes the control software for the control modes. Redundant UPS sources power the FWCS digital controllers and process measurement equipment. No single power source or single power supply failure results in the loss of FWCS functions. Each parallel processing channel has independent inputs and outputs.

The FTDC self-test and online diagnostic test features are capable of identifying and isolating failures of process sensors, I/O cards, power buses, power supplies, processors, and intermediate processor communication paths. These features identify the presence of a fault and determine the location of the failure down to the module level.

Further, in the event that the failure of the triple redundant FWCS causes the RPV water level to drop, when the water level reaches Level 2, the high pressure makeup function of the CRD system initiates as a diverse means. During normal operation, feedwater flow is delivered to the RPV through three RFPs which operate in parallel. In normal operation, the fourth RFP is in standby mode and starts automatically if any operating feedwater pump trips while at power. Also, the fourth RFP can be set in manual mode or can be removed from service for maintenance. The RPV water level, steam flow, and feedwater flow use multiple signals that are displayed and alarmed in the MCR.

During normal operation, the FWCS sends three speed-demand signals, each of which reflects a voted FWCS processor output, to each feed pump ASD. The ASD performs a mid-value vote and uses it to control the speed/frequency of the feed pump motor. The mid-value vote is also returned to the FWCS as an analog input and compared with the speed demands sent by the FWCS. If an FTDC channel detects a discrepancy between the field voter output and the FTDC channel output, a "lockup" signal is sent to a "lockup" voter and an alarm is activated in the MCR.

Implementation of the two control schemes in physically different cabinets and logic processors precludes a combined feedwater temperature change and feedwater flow/reactor water level change caused by controller failure.

Based on the above, the staff finds that the FWCS design limits the potential for inadvertent actuation.

7.7.3.2.2 Evaluation of Feedwater Control System Compliance with Regulations

Section 7.7.0.4 of this report discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for the control systems identified in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the FWCS.

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the FWCS. The staff reviewed DCD Tier 2, Revision 9, Section 7.7.3, and DCD Tier 1, Revision 9, Section 2.2.3, in accordance with SRP Sections 7.7 and 14.3.5, to verify that 10 CFR 52.47(b)(1) is adequately addressed for the FWCS. DCD Tier 1, Section 2.2.3, documents the FWCS ITAAC requirements. While the FWCS has no DAC, the FWCS methods and functions of controlling the feedwater and feedwater temperature, the general equipment and modes of control, and the actuation initiators of protective actions are specified. The staff noted that DCD Tier 1, Revision 9, Table 2.2.3-1, states, "FWCS is a triple-redundant, fault tolerant digital controller (FTDC)." Based on information in DCD Tier 1, Revision 9; DCD Tier 2, Revision 9, Chapter 7; information discussed herein; and identified FWCS I&C and their verification in the ITAAC, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed.

To determine compliance with GDC 10, 13, 15, 19, 28, and 29, the staff reviewed DCD Tier 2, Revision 9, Section 7.7.3, to verify that the applicable GDC specified in SRP Section 7.7 are adequately addressed for the FWCS as a nonsafety system of the ESBWR. DCD Tier 2, Revision 9, Section 7.7.3.3.2, states that the FWCS complies with GDC 13 and 19.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the FWCS. SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. In DCD Tier 2, Revision 9, Chapter 15, Tables 15.1-5 and 15.1-6 identify FWCS actuations and other actions that reduce the need for the actuation of protection and safety systems to mitigate AOOs. DCD Tier 2, Revision 9, Section 7.7.3, includes corresponding actions in the design bases of the FWCS to maintain the reactor core, reactor coolant system, and the reactivity limits within appropriate margins and to mitigate AOOs. Examples of such design features are discussed below.

The FWCS controls the RPV water level between Level 8 and Level 3 during normal operation. Consistent with DCD Tier 2, Revision 9, Table 15.1-5, the FWCS mitigates AOOs such as Inadvertent Isolation Condenser Initiation and Runout of One Feedwater Pump events by reducing the output of the remaining feedwater pumps to compensate for the excessive output of the failed pump and to maintain appropriate reactor water level. The FWCS controls feedwater temperature around a setpoint and neither the operator nor the automation system can change the setpoint faster than an allowable rate (nominally 55.6 degrees C [100 degrees F] per hour). The FWCS design does not allow a feedwater temperature decrease when the reactor thermal power exceeds 100 percent of rated thermal power. The FWCS initiates a runback of the feedwater pump demand to zero and the closing of the LFCV and RWCU/SDC OBCV when an ATWS trip signal is received. DCD Tier 1, Revision 9, Section 2.2.3, includes the ITAAC for the applicant to verify that the as-built FWCS implements these actions. Accordingly, based on identified FWCS actions and their verification in the ITAAC, the staff finds that the requirements of GDC 10 are adequately addressed for the FWCS.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the FWCS, for AOOs, and for accident conditions, as appropriate, to help assure adequate safety. For the FWCS, the I&C includes instrumentation for parameters and controls that affect reactivity and RPV water level. The staff reviewed the plant transient response to normal load changes and AOOs such as Loss of Feedwater Heating, Feedwater Controller Failure-Maximum Flow Demand, LOFWH With Failure of SCRRI and SRI, and Loss of All Feedwater. The staff concludes that the FWCS is capable of maintaining system variables within prescribed operating ranges and implementing contingency actions if such variables reach limiting conditions. DCD Tier 2, Revision 9, Sections 7.7.3.2 and 10.4.7.5, specify status information provided by and for the FWCS, including status data on feedwater flow rate, steam flow rate, RFP discharge flow rates, and monitoring of the temperature and temperature difference between feedwater lines A and B (which, if excessive, provides an indication to the operator). DCD Tier 2, Revision 9, Section 7.7.3.2.2, specifies manual and automatic reactor water level controls. DCD Tier 2, Revision 9, Section 7.7.3.2.3, specifies manual and automatic reactor water temperature controls. DCD Tier 2, Revision 9, Section 7.7.3.3.2, indicates compliance with GDC 13. DCD Tier 1, Revision 9, Section 2.2.3, includes the ITAAC for the applicant to verify that the as-built FWCS implemented the required automatic functions and operator reactor water level and temperature controls. Accordingly, based on identified FWCS I&C and their verification in the ITAAC, the staff finds that the requirements of GDC 13 are adequately addressed for the FWCS.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the FWCS. SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate the I&C system contributions to design margin for reactor coolant systems. The staff reviewed DCD Tier 2, Revision 9, Section 7.7.3, and DCD Tier 1, Revision 9, Section 2.2.3, for the features of manual and automatic control described in the FWCS that facilitate the capability to maintain plant variables within prescribed operating limits and over their anticipated ranges for normal operation, for AOOs, and for accident conditions, as appropriate, to assure adequate safety. These include examples such as the automatic and manual controls for the RFPs, RPV water level measurement instrumentation, steam flow measurement instrumentation, and feedwater flow measurement instrumentation. Accordingly, based on identified FWCS I&C and their verification in the ITAAC, the staff finds that the requirements of GDC 15 are adequately addressed for the FWCS.

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the FWCS. In Section 7.1.1.3.6 the staff evaluated that GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should evaluate if there exists I&C available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.3.3.2, specifies that the FWCS conforms to GDC 19. DCD Tier 2, Revision 9, Section 7.7.3.5, describes parameters provided by the FWCS to the MCR. Examples of major measurement include RPV water level, main steam flow rate, and feedwater flow rate. DCD Tier 2, Revision 9, Section 7.7.3.2.2, specifies manual and automatic reactor water level controls. DCD Tier 2, Revision 9, Section 7.7.3.2.3, specifies manual and automatic reactor water temperature controls. DCD Tier 1, Revision 9, Section 2.2.3, includes the ITAAC for the applicant to verify that the as-built FWCS implements the required automatic functions and operator reactor water level and temperature controls. Based on the above, including the features for manual and automatic control described in DCD Tier 2, Revision 9, Section 7.7.3, the staff finds that the requirements of GDC 19 are adequately addressed for the FWCS.

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. The staff evaluated whether GDC 28 is adequately addressed for the FWCS. SRP Appendix 7.1-A states that GDC 28 imposes functional requirements on I&C interlock and control systems to the extent they are provided to limit reactivity increases to prevent or limit the effect of reactivity accidents. DCD Tier 2, Revision 9, Section 7.7.3.3.2, does not include GDC 28. However, DCD Tier 2, Revision 9, Section 7.7.3.2.3, summarizes the feedwater temperature control features that limit the potential amount and rate of reactivity increase. These features include (1) neither the operator nor the automation system can change the setpoint faster than an allowable rate (nominally 55.6 degrees C [100 degrees F] per hour), (2) neither the operator nor the automation system can input a setpoint outside the area allowed by the reactor power versus feedwater temperature operating map (power- feedwater temperature map), and (3) the feedwater temperature controller is unable to decrease feedwater temperature if the reactor thermal power is equal to or greater than 100 percent. Similarly, DCD Tier 2, Revision 9, Section 7.7.3.2.2, summarizes the feedwater level controls that limit the potential amount and rate of reactivity increase. For example, if the reactor water level reaches Level 8, the FWCS simultaneously activates an MCR alarm, sends a zero-speed demand signal to the feed pump ASDs, and trips the main turbine. These and other functions of the design performed by the FWCS address GDC 28 and reduce the need for the actuation of safety systems. DCD Tier 1, Revision 9, Section 2.2.3, includes the ITAAC for the applicant to verify that the as-built FWCS implements these actions. Accordingly, based on the identified FWCS actions and their verification in the ITAAC, the staff finds that the requirements of GDC 28 are adequately addressed for the FWCS.

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed. SRP Appendix 7.1-A identifies that GDC 29 is addressed by conformance, as applicable, to GDC 20 thru 25 and GDC 28. Since the FWCS is a reactivity control system and not a protection system, GDC 20-23 and 25, which are requirements for protection systems, do not apply to the FWCS. Accordingly, GDC 29 is addressed by conformance as applicable to GDC 24 and 28. Section 7.7.0.4 of this report evaluates conformance of N-DCIS control systems, including the FWCS, to GDC 24. Conformance of the FWCS to GDC 28 is evaluated above. Further, DCD Tier 2, Revision 9, Section 7.7.3.2.1, states that each function of the FWCS is implemented on its own dedicated set of the triple redundant FTDCs, including power supplies and input/output signals. The controller is designed for a mean time to failure of no less than 1,000 years, according to the applicant. Each set of FTDCs consists of three parallel processing channel controllers, each containing the hardware and software for execution of the control algorithms. Each FTDC channel executes the control software for the control modes. Based on the information above, and adequate consideration of the requirements of GDC 24 and 28 for the FWCS, the staff finds that the requirements of GDC 29 are adequately addressed for the FWCS.

Therefore, the staff finds that the FWCS adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 above for a nonsafety system and that there is reasonable assurance this system will be able to accomplish its designed function in a reliable manner, when built and tested according to DCD Tier 2, Revision 9, and the DCD Tier 1, Revision 9, ITAAC.

7.7.3.3 *Conclusion*

Based on the above, staff concludes that (1) there is reasonable assurance that the FWCS conforms to the applicable requirements, which include GDC 1, 10, 13, 15, 19, 24, 28, and 29; 10 CFR 52.47(b)(1); 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h); (2) adequate high level functional requirements are identified, and (3) sufficient ITAAC are included in DCD Tier 1, Revision 9, to verify that the design is completed in compliance with the applicable requirements.

7.7.4 Plant Automation System

7.7.4.1 *Summary of Technical Information*

As described in DCD Tier 2, Revision 9, Section 7.7.4, the nonsafety PAS provides the capability for supervisory control of the entire plant. It does this by supplying setpoint commands to independent nonsafety automatic control systems, such as the FWCS and RC&IS. The PAS provides supervisory controls that regulate reactivity during criticality control, provides heatup and pressurization control, regulates reactor power, controls turbine and generator output, controls secondary nonsafety systems, and provides reactor startup and shutdown controls. The functions of the PAS are accomplished by suitable algorithms for different phases of reactor operation, which include approaches to criticality, heatup, reactor power increase, automatic load following, reactor power decrease, and shutdown. The triple redundant FTDC and redundant system controllers perform the PAS control functional logic.

The N-DCIS accepts one-way communication from the Q-DCIS so that the safety information can be monitored, archived, and alarmed seamlessly with the N-DCIS data. Through the N-DCIS, the PAS receives input from the safety NMS and RPS. Through the N-DCIS, the PAS receives input from the nonsafety RC&IS, SB&PC system, PAS, RWCU/SDC, and the TGCS. The PAS sends output demand request signals to the RC&IS to position the control rods, to the SB&PC for pressure setpoints, and to the TGCS for load following operation. DCD Tier 2, Revision 9, Figure 7.7-4, provides a simplified functional block diagram of the PAS.

The PAS interfaces with the operator's control console in the MCR to perform its designed functions. From the operator's control console for automatic plant startup, power operation, and shutdown functions, the operator uses the PAS to issue supervisory control commands to nonsafety systems. The operator also uses the PAS to adjust setpoints of lower level controllers to support automation of the normal plant startup, shutdown, and power range operations.

In the automatic mode, the PAS also issues command signals to the turbine master controller, which contains appropriate algorithms for automated sequences of the main turbine and related auxiliary systems. The PAS presents the operator with a series of breakpoint controls on the main control console nonsafety VDUs for a prescribed plant operation sequence.

When all of the prerequisites are satisfied for a prescribed breakpoint in a control sequence, the PAS provides the status and requests permission from the operator to proceed. Upon operator acceptance, the prescribed control sequence is initiated or continued. The PAS then initiates demand signals to various system controllers to carry out the predefined control functions. For non-automated operations required during normal startup or shutdown (such as a change of reactor mode switch status), automatic prompts are provided. Automated operations continue after the prompted actions are completed manually. The PAS performs the functions associated with reactor power control.

For reactor power control, the PAS contains algorithms that can change reactor power by control rod motions. A prescribed control rod sequence is followed when manipulating control rods for reactor criticality, heatup, power changes, and automatic load following. Each of these functions has its own algorithm to achieve its designed objective. When the reactor power control is to be done by feedwater temperature change, the PAS can provide the FWCS feedwater control setpoints to permit reactor power maneuvering without control rod motion. During automatic load-follow operation, the PAS interfaces with the TGCS to coordinate main turbine and reactor power changes for stable operation and performance.

7.7.4.2 *Staff Evaluation*

7.7.4.2.1 Evaluation of PAS Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations outlined in SRP Section 7.7 and discusses the attributes that are common to the control systems of DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those major design considerations and information unique to the PAS.

(1) Design Basis

The PAS has no safety design basis. This system is designed so that it does not affect the functionalities of safety systems. The PAS does have a power generation (nonsafety) design basis. The nonsafety design basis of the PAS is to provide supervisory (automatic and semi-automatic) control that regulates reactivity during criticality control, provides heatup and pressurization control, regulates power, controls turbine and generator output, controls secondary-related nonsafety systems, and provides reactor startup and shutdown control. The functions of the PAS are accomplished by suitable algorithms for different phases of reactor operation. Through the N-DCIS, the PAS receives input from the safety systems including the NMS and the RPS. Through the N-DCIS, the PAS receives input from the nonsafety systems, including the RC&IS, SB&PC system, FWCS, RWCU/SDC, and TGCS. The algorithms then calculate appropriate setpoints that are provided back to various control systems including the (1) RC&IS to position the control rods, (2) SB&PC system for pressure setpoints, and (3) TGCS for load-follow operation. For example, to control reactor power by feedwater temperature change, the PAS provides the feedwater temperature control with a new setpoint to allow reactor power maneuvering without moving control rods. Based on information reviewed in DCD Tier 2, Revision 9, Section 7.7.4; DCD Tier 2, Revision 9, Chapter 15; and the evaluations that follow, the staff finds that the PAS includes the necessary features for automatic control and semi-automatic control of process variables within prescribed operating limits.

(2) Effects of Control System Operation on Accidents

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.4 and Chapter 15, with regard to the effects of PAS operation on accidents. The PAS does not directly control the physical equipment, such as pumps, valves, and control rods, but receives inputs from systems that control the physical equipment and provides setpoints for automatic and semi-automatic plant operations. The normal mode of operation of the PAS is automatic. If any system or component conditions are abnormal during execution of the prescribed sequences, the PAS automatically switches into the manual mode. With the PAS in manual mode, any in-progress operation stops and alarms activate in the MCR. Further, with the PAS in manual mode, the operator can manipulate control rods through the normal controls. A failure of the PAS does not prevent manual control of reactor power and does not prevent safe shutdown of the reactor. Inaction by the PAS when in automatic mode is effectively an input failure or an I&C component failure, which are discussed below under the effects of control system failures caused by accidents. Based on the above, and as described in DCD Tier 2, Revision 9, Section 7.7.4 and Tables 15.1-5 and 15.1-6, the staff finds that the safety analysis presented in DCD Tier 2, Revision 9, Chapter 15, considers the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOOs.

(3) Effects of Control System Failures

The staff reviewed DCD Tier 2, Revision 9, Chapters 15 and Section 7.7.4 with regard to the effects of PAS failures. The PAS does not control physical equipment directly, but provides control functions through setpoint changes in systems that control the physical equipment. The transients presented in DCD Tier 2, Revision 9, Tables 15.1-5, 15.1-6, and 15.1-7, are bounding for PAS failures. A PAS failure could result from bad input or component failure within the PAS. For example, assume a failure in the PAS provided a setpoint to the FWCS that caused a significant increase in feedwater flow. This would be bounded by the DCD Tier 2, Revision 9, Table 15.1-6, transient, "Feedwater Controller Failure-Maximum Flow Demand." An incorrect setpoint failure of the PAS does not prevent manual control of reactor power, and does not prevent safe shutdown of the reactor. The staff verified that the failure of any PAS system component for control systems does not cause plant conditions more severe than those described in the analysis of AOOs in DCD Tier 2, Revision 9, Chapter 15. Based on the information reviewed in DCD Tier 1, Revision 9, and DCD Tier 2, Revision 9, Section 7.7.4, and Chapter 15, the staff finds that the occurrences and events discussed in DCD Tier 2, Revision 9, Chapter 15 are consistent with failures of the PAS; therefore, the staff finds that failure of the PAS does not cause plant conditions more severe than those described in DCD Tier 2, Revision 9, Chapter 15.

In RAIs 7.7-12 and 7.7-12, Supplement 1, the staff requested that the applicant provide analyses that evaluate the effects of PAS control system failures. RAI 7.7-12 was being tracked as an open item in the SER with open items. In its response, the applicant revised the DCD Tier 2, Section 7.7 to reference DCD Tier 2, Chapter 15 analyses of specific events that evaluate the effects of control systems failures. The applicant also stated that, in the unlikely event that the failure of the triple redundant master controllers and duplicate system controllers cause the PAS to issue incorrect setpoints, the expected and abnormal transients and accident events analyzed in DCD Tier 2, Chapter 15 bound the effects of the PAS failures as discussed above. The DCD Tier 2, Chapter 15 events that analyze the effects of the PAS failures are as follows: (1) Sections 15.2.3.1, 15.2.3.2, 15.3.8, and 15.3.9 bound the effects of failures of the RC&IS controls, (2) Sections 15.2.4.2, 15.3.1, and 15.3.2 bound the effects of failures of FWCS controls, and (3) Sections 15.2.5.1, 15.3.3, 15.3.4, 15.3.2, and 15.3.6 bound the effects of failures of the SB&PC system controls. The staff finds that the response is acceptable since the applicant identified the DCD Tier 2, Revision 9, Chapter 15 analyses that bound the failures of the PAS. Based on the applicant's response, RAI 7.7-12 regarding the PAS is resolved.

(4) Effects of Control System Failures Caused by Accidents

The staff reviewed DCD Tier 2, Revision 9, Chapters 15 and Section 7.7.4, regarding the effects of PAS failures caused by accidents. The PAS instrumentation includes (1) MCR instrumentation for the man-machine interface, (2) hardware and software for input/output interfaces and controller functions, and (3) direct non-multiplexed sensor inputs needed by the system. Except for some inputs, the PAS equipment is designed for a mild environment. Accidents could potentially damage the sensors and transmitters providing input to the PAS. Fire or earthquake potentially could cause one or more functions to fail and damage the electronic equipment causing a PAS function to fail. The potential for such effects is significantly reduced through D3 in the NBS(N) sensors providing input, environmental equipment qualification, triple redundant master controllers and duplicate system controllers modules, and a fire protection design. Implementation of the two control schemes in physically different cabinets and logic processors precludes a combined feedwater temperature change and feedwater flow/reactor water level change caused by controller failure. Based on the above, the evaluation of effects of control system failures above, and a review of accident consequences and effects of AOOs on the PAS, the staff finds that the consequential effects of AOOs and accidents do not lead to PAS failures that would result in consequences more severe than those described in DCD Tier 2, Revision 9, Chapter 15. Based on the above, the staff finds that the effects of PAS failures caused by accidents are adequately addressed.

(5) Potential for Inadvertent Actuation

The staff reviewed DCD 2, Revision 9, Section 7.7.4, to identify design measures that limit the potential for inadvertent actuation. Examples of such design features follow.

The PAS is designed so that it does not affect the functionalities of the safety systems. Through the N-DCIS, the PAS receives input from the safety NMS and RPS. The N-DCIS accepts one-way communication from the Q-DCIS so that the safety information can be monitored, archived, and alarmed seamlessly within the N-DCIS.

The PAS hardware comprises triple redundant master controllers and duplicate system controllers. The normal mode of operation of the PAS is automatic. This supports a decrease in the potential for inadvertent actuation by reducing the potential for human error. If any system or component conditions are abnormal during execution of the prescribed sequences, the PAS automatically switches into the manual mode. With the PAS in manual mode, any operation in progress stops, and alarms are activated in the MCR. Also, with the PAS in manual mode, the operator can manipulate control rods through the normal controls. The FTDC input and output communication interfaces function continuously during normal power operation. Abnormal functioning of these components can be detected during operation. In addition, the FTDC is equipped with self-test and on-line diagnostic capabilities for identifying and isolating failures of input/output signals, buses, power supplies, processors, and inter-processor communications. These functions can be performed without interrupting the normal control operation of the PAS.

Based on the above, the staff finds that the PAS design limits the potential for inadvertent actuation.

7.7.4.2.2 Evaluation of Plant Automation System Compliance with Regulations

Section 7.7.0.4 of this report lists and discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for the control systems identified in DCD Tier 2, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the PAS.

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the PAS. DCD Tier 1 does not provide ITAAC for the PAS. The PAS is not only nonsafety, it does not have direct control over the physical position of control rods, control feedwater flow, or control steam flow rate. In addition, the PAS is not required by any regulation. Accordingly, based on a review of DCD Tier 2, Revision 9, Section 7.7.4, and SRP Sections 14.3 and 14.3.5, the staff finds that not providing ITAAC for PAS is acceptable.

To determine compliance with GDC 10, 13, 15, 19, 28, and 29, the staff reviewed DCD Tier 2, Revision 9, Section 7.7.4, to verify that the applicable GDC are adequately addressed for the PAS as a nonsafety system. DCD Tier 2, Revision 9, Section 7.7.4.3.2, states that the PAS conforms to GDC 13 and 19.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the PAS. SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. The PAS functions as a supervisor system with no direct control of physical equipment that controls reactivity or power. The PAS receives input from the following major nonsafety systems: the RC&IS (Section 7.7.2), SB&PC system (Section 7.7.5), FWCS (Section 7.7.3), RWCU/SDC (Section 7.4.3), and TGCS. The PAS sends output demand request signals to the RC&IS to position the control rods, to the SB&PC system for pressure setpoints, and to the TGCS for load following operation. Thus, since the PAS does not have direct control, the staff finds that GDC 10 does not apply to the PAS.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the PAS. DCD Tier 2, Revision 9, Section 7.7.4.2, identifies I&C provided to monitor, control, and maintain the variables over their anticipated ranges for normal operation during the automatic and non-automatic (semi-automatic) operation of the PAS and while it is in the manual mode. The PAS instrumentation continues to receive input when in manual mode, but any in-progress operation stops and alarms in the MCR are initiated. Another example is that the PAS has instrumentation to interface with the operator's control console to aid in its designed functions. From the operator's control console for automatic plant startup, power operation, and shutdown functions, the operator uses the PAS to issue supervisory control commands to nonsafety systems. The operator also uses the PAS to adjust setpoints of lower level controllers. The PAS provides supervisory direction to the actual controlling functions. The PAS presents the operator with a series of breakpoint controls on the main control console through nonsafety VDUs for a prescribed plant operation sequence. After all prerequisites are satisfied for a prescribed breakpoint on a control sequence, the PAS provides a status and a permissive (i.e., a control system request to the reactor operator for permission to proceed) is requested. The prescribed control sequence is initiated only following operator acceptance. If any system or component conditions are abnormal during execution of the prescribed sequences, the PAS automatically switches into the manual mode. DCD Tier 2, Revision 9, Section 7.7.4.3.2, indicates conformance to GDC 13. Accordingly, based on a review of DCD Tier 2, Revision 9, Chapter 15 and Section 7.7.4, and the applicant’s commitment to provide instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions, as appropriate, to ensure adequate safety and to provide appropriate controls to maintain these variables and systems within prescribed operating ranges, the staff finds that the requirements of GDC 13 are adequately addressed for the PAS.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the PAS. SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate the I&C system contributions to design margin for reactor coolant systems. DCD Tier 2, Revision 9, Section 7.7.4.2, identifies the PAS I&C provided to monitor, control, and maintain the variables over their anticipated ranges. The output demand request signals from the PAS are sent to the RC&IS to position the control rods, to the SB&PC system for pressure setpoints, and to the TGCS for load following operation. The FWCS controls the RPV water level while providing the main reactor coolant, but it does not need setpoints from the PAS. Since the PAS does not directly control parameters that contribute the design margins for the RCPB, GDC 15 does not apply to the PAS.

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the PAS. In Section 7.1.1.3.6 the staff evaluated that GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should be evaluated if there exists I&C available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.4.3.2, specifies that the PAS conforms to GDC 19. DCD Tier 2, Revision 9, Section 7.7.4.2, describes the PAS interface with the operator’s console to perform PAS designated functions. An example includes parameters provided by the PAS to the MCR. From the operator's control console for automatic plant startup, power operation, and shutdown functions, the operator uses the PAS to issue supervisory control commands to nonsafety systems. The operator also uses the PAS to adjust setpoints of lower level controllers to support automation of the normal plant startup, shutdown, and power range operations. DCD Tier 2, Revision 9, Section 7.7.4.2.2, specifies manual and automatic PAS controls. Based on the above, including the features for manual and automatic control described in DCD Tier 2, Revision 9, Section 7.7.4, the staff finds that the requirements of GDC 19 are adequately addressed for the PAS.

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. The staff evaluated whether GDC 28 is adequately addressed for the PAS. SRP Appendix 7.1-A states that GDC 28 imposes functional requirements on I&C interlock and control systems to the extent they are provided to limit reactivity increases to prevent or limit the effect of reactivity accidents. DCD Tier 2, Revision 9, Section 7.7.4.3.2, does not include GDC 28. DCD Tier 2, Revision 9, Section 7.7.4.2, describes the output of setpoints that specifically control reactivity and thus reactor power. The algorithms have appropriate limits on control rod motion when manipulating control rods for reactor criticality, heatup, power changes, and automatic load following. Other algorithms provide setpoints for the reactivity control by feedwater temperature change with appropriate restrictions and without moving control rods. In combination, the two reactor power control methods form a sequential step-by-step power maneuvering strategy for the control rod pattern/movement and feedwater temperature change. Based on the above, as well as information from DCD Tier 2, Revision 9, Section 7.7.4, the staff finds that the requirements of GDC 28 are adequately addressed.

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed. SRP Appendix 7.1-A identifies that GDC 29 is addressed by conformance, as applicable, to GDC 20-25 and GDC 28. Since the PAS is a reactivity control system and not a protection system, GDC 20-23 and 25, which are requirements for protection systems, do not apply to the PAS. Accordingly, GDC 29 is addressed by conformance as applicable to GDC 24 and 28. Section 7.7.0.4 of this report evaluates conformance of N-DCIS control systems, including the PAS, to GDC 24. Conformance of the PAS to GDC 28 is evaluated above. Further, DCD Tier 2, Revision 9, Section 7.7.4.5, states that the PAS hardware comprises triple redundant master controllers and duplicate system controllers. In support of a high probability of accomplishing the PAS design function, the controllers are FTDC with input and output communications interfaces continuously functioning during normal power operation. The FTDC have on-line self-test and diagnostics which can be performed without interrupting the PAS normal control operation. Based on the information above, and adequate consideration of the requirements of GDC 24 and 28, the staff finds that the requirements of GDC 29 are adequately addressed for the PAS.

Therefore, the staff finds the PAS adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 above for a nonsafety system and that there is reasonable assurance that this system will be able to accomplish its design function in a reliable manner when built and tested according to DCD Tier 2, Revision 9.

7.7.4.3 *Conclusion*

Based on the above, the staff concludes that (1) there is reasonable assurance that the PAS conforms to the applicable requirements, which include GDC 1, 13, 19, 24, 28, and 29; 10 CFR 52.47(b)(1); 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h); and (2) adequate high level functional requirements are identified in compliance with the applicable requirements.

7.7.5 Steam Bypass and Pressure Control System

7.7.5.1 *Summary of Technical Information*

The SB&PC system controls reactor pressure during plant startup, power generation, and shutdown modes of operation.

As described in DCD Tier 2, Revision 9, Section 7.7.5.2, the control of reactor pressure is accomplished through control of the TCVs through the TGCS and TBVs, so as to minimize susceptibility to reactor trip, turbine-generator trip, main steam isolation, and SRV opening. Triple redundant FTDCs using feedback signals from RPV dome pressure sensors generate command signals for the TBVs and pressure regulation demand signals used by the TGCS to generate demand signals for the TCVs. For normal operation, the TCVs regulate reactor pressure. Whenever the total steam flow demand from the SB&PC system exceeds the effective TCV steam flow demand, the SB&PC system sends the excess steam flow directly to the main condenser through the TBVs. The ability of the plant to load-follow the grid-system demands is accomplished by the aid of control rod actions. In response to the resulting steam production demand changes, the SB&PC system adjusts the demand signals sent to the TGCS so that the TGCS adjusts the TCVs to accept the control steam output change, thereby controlling pressure. DCD Tier 2, Revision 9, Section 7.7.5.6, describes the major instrument interfaces with the SB&PC system. The SB&PC system also has the capability to start the auxiliary boiler and command the auxiliary boiler to adjust steam production rate upon MSIV closure conditions, as required.

7.7.5.2 *Staff Evaluation*

7.7.5.2.1 Evaluation of Steam Bypass and Pressure Control Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations identified in SRP Section 7.7 and discusses the attributes that are common to the control systems described in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those major design considerations and information unique to the SB&PC system.

(1) Design Basis

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.5, and the design basis of the SB&PC system. The SB&PC system is not required to operate during or after any DBA; therefore, the SB&PC system has no safety design basis. The SB&PC system is required for power generation because it controls reactor pressure during plant startup, power operation, and shutdown modes. The SB&PC system design objective is to enable a fast and stable response to system pressure disturbances, and to pressure setpoint changes over the operating range. DCD Tier 2, Revision 9, Sections 7.7.5.2.2, 7.7.5.2.3, 7.7.5.4, and 7.7.5.6.10, summarize normal, abnormal and special operational features and functions. For example, during events that lead to a reactor trip, the SB&PC system functions to stabilize the system pressure, thus aiding the FWCS feedwater level control in maintaining RPV water level. Based on information reviewed in DCD Tier 1, Revision 9, Section 2.2.9, and DCD Tier 2, Revision 9, Section 7.7.5 and Chapter 15, the staff finds that the SB&PC system includes the necessary features for manual and automatic control of process variables within prescribed operating limits.

(2) Effects of Control System Operation on Accidents

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.5 and Chapter 15 regarding the effects of control system operation on accidents. The staff verified that the safety analysis considers the effects of both SB&PC system action and inaction in assessing the transient response of the plant for accidents and AOOs. DCD Tier 2, Revision 9, Sections 7.7.5.1 and 7.7.5.2, summarize the objective of the SB&PC. The SB&PC system mission is to enable a fast and stable response to system pressure disturbances, and to pressure setpoint changes over the operating range. This is done by modulating TCVs by providing signals to the TGCS and by modulating TBVs for controlling reactor pressure. In addition, the design objective of the SB&PC system is to discharge reactor steam directly to the main condenser to regulate reactor pressure whenever the main turbine cannot use all of the steam generated by the reactor. The SB&C system mitigates AOOs by (1) stabilizing system pressure, thereby aiding the feedwater level control systems in maintaining RPV water level, and (2) operating with other reactor control systems to avoid reactor trip after significant plant disturbances, such as loss of one feedwater pump, inadvertent opening of an SRV or TBV, main turbine stop/control valve surveillance testing, and MSIV testing. SB&PC system inaction implies a component failure in the SB&PC system logic and is addressed under effects of control system failures below. Based on the information reviewed in DCD Tier 2, Revision 9, Section 7.7.5 and Chapter 15, and as described under effects of control system failures below, the staff finds that the safety analysis considers the effects of both control system action and inaction in assessing the transient response of the plant for accidents and AOOs.

(3) Effects of Control System Failures

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.2 and Chapter 15 regarding the effects of the SB&PC system failures. DCD Tier 2, Revision 9, Tables 15.1-5, 15.1-6, and 15.1-7 identify several occurrences and events related to the SB&PC system that bound the SB&PC system component failure. These transients are (1) Pressure Regulator Failure Opening All Turbine Control Valves and Bypass Valves, (2) Pressure Regulator Failure-Closure of All Turbine Control Valves and Bypass Values, (3) Generator Load Rejection with total Turbine Bypass Failure (at High Power), and (4) Turbine Trip with Total TBV Failure (at High Power). Based on information reviewed in DCD Tier 2, Revision 9, Section 7.7.5; DCD Tier 2, Revision 9, Chapter 15; and evaluations herein, the staff finds that the failure of the SB&PC system does not cause plant conditions more severe than those described in DCD Tier 2, Revision 9, Chapter 15. In RAI 7.7-12, the staff requested that the applicant provide analyses that evaluate the effects of control systems failures. RAI 7.7-12 was being tracked as an open item in the SER with open items. In its response, the applicant revised the DCD Tier 2, Section 7.7, to reference DCD Tier 2, Chapter 15, analyses of specific events that evaluate the effects of control systems failures. As described above, the expected and abnormal transients and accident events analyzed in DCD Tier 2, Sections 15.2.5.1, 15.3.3, 15.3.4, 15.3.5, and 15.3.6 bound the effects of the SB&PC system failures. The staff finds that the response is acceptable since the applicant identified the DCD Tier 2, Revision 9, Chapter 15 analyses that bound the failures of the SB&PC. Based on the applicant's response, RAI 7.7-12 regarding the SB&PC is resolved.

(4) Effects of Control System Failures Caused by Accidents

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.5 and Chapter 15, regarding the effects of the SB&PC system failures caused by accidents. The SB&PC system instrumentation includes the triple redundant FTDC panel mounted in the main control room back panel (MCRBP), communication interfaces to the N-DCIS cabinets, and inputs from other systems. Accidents could potentially damage the sensors, transmitters, or communication providing input to the SB&PC. Fire or earthquake potentially could cause one or more functions to fail and damage the electronic equipment, causing an SB&PC system function to fail. The potential for such effects is significantly reduced through D3 design features, SB&PC system environmental EQ, triple redundant controller modules in different cabinets, and a fire protection design. Based on the above, the discussion of the effects of control system failures, and a review of accident consequences and the effects of AOOs on the SB&PC, the staff finds that the consequential effects of AOOs and accidents do not lead to SB&PC failures that would result in consequences more severe than those described in DCD Tier 2, Revision 9, Tables 15.1-5 and 15.1-6. Based on the above, the staff finds that the effects of SB&PC failures caused by accidents are adequately addressed.

(5) Potential for Inadvertent Actuation

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.5, to identify design measures that limit the potential for inadvertent actuation. Examples of such design features are described below.

The SB&PC system is implemented on the triple redundant FTDCs. The SB&PC system has three redundant nonsafety ac UPSs. The SB&PC system panel is designed so that a loss of one power supply or incoming power source does not affect the SB&PC system functional operation and thus plant operation. Triple redundant FTDCs perform the SB&PC system functional logic and process control functions. Because of the triple redundancy, it is possible to lose one complete processing channel without affecting the system function. One channel may be taken out of service for maintenance, repair, or module replacement while the system is online. In power operation, loss of more than one of the triply redundant FTDCs will trip the main turbine. Controls and valve positions are designed so that steam flow is shut off when the control system electrical power or hydraulic system pressure is lost.

The SB&PC system helps avoid inadvertent actuation of the RPS. During normal operational plant maneuvers, the SB&PC system provides responsive, stable performance to minimize RPV water level and neutron flux transients. The SB&PC system provides for automatic control of the reactor pressure during plant startup and heatup. The SB&PC system is also designed to operate with other reactor control systems to avoid a reactor trip after significant plant disturbances. Examples of such disturbances are loss of one feedwater pump, inadvertent opening of SRVs or TBVs, main turbine stop/control valve surveillance testing, and steamline isolation valves testing. To protect the condenser, the SB&PC system inhibits opening of the TBVs when it detects high condenser pressure.

Based on the above, the staff finds that the SB&PC system design limits the potential for inadvertent actuation.

7.7.5.2.2 Evaluation of Steam Bypass and Pressure Control Compliance with Regulations

Section 7.7.0.4 of this report lists and discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for the control systems identified in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the SB&PC system.

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are adequately addressed for the SB&PC system. The staff reviewed DCD Tier 2, Revision 9, Section 7.7.5 and DCD Tier 1, Revision 9, Section 2.2.9, in accordance with SRP Sections 7.7 and 14.3.5, to verify that 10 CFR 52.47(b)(1) is adequately addressed for the SB&PC. DCD Tier 1, Section 2.2.9, documents the SB&PC system ITAAC requirements. The SB&PC system methods and functions of controlling reactor pressure are specified. The staff noted that DCD Tier 1, Section 2.2.9, states that the SB&PC system uses triple-redundant FTDCs. Based on information in DCD Tier 1, Section 2.2.9, and DCD Tier 2, Chapter 7.7.5; information discussed herein; and the identified SB&PC system I&C and their verification in the ITAAC, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed.

To determine compliance with GDC 10, 13, 15, 19, 28, and 29, the staff reviewed DCD Tier 2, Revision 9, Section 7.7.5, to verify that the applicable GDC specified in SRP Section 7.7 are adequately addressed for the SB&PC system as a nonsafety system. DCD Tier 2, Revision 9, Section 7.7.5.3.2, states that the SB&PC system conforms to GDC 13 and 19, and 24.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the SB&PC system. SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. DCD Tier 2, Revision 9, Chapter 15 and Tables 15.1-5 and 15.1-6, identify actuations and other actions that reduce the need for the actuation of protection and safety systems to mitigate AOOs. DCD Tier 2, Revision 9, Section 7.7.5, includes corresponding actions in the design bases of the SB&PC system to maintain the RPV pressure, reactor coolant system, and reactivity limits within appropriate margins and to mitigate AOOs. The specific objective of the SB&PC system is to control reactor pressure during plant startup, power generation, and shutdown modes of operation. Further, for normal operation, the TCVs regulate reactor pressure. However, whenever the total steam flow demand from the SB&PC system exceeds the effective TCV steam flow demand, the SB&PC system sends the excess steam flow directly to the main condenser through the TBVs. The ability of the SB&PC system to control reactor pressure supports the FWCS RPV level and provides significant improvement in pressure and RPV water level control during transients. DCD Tier 1, Revision 9, Section 2.2.9, includes the ITAAC for the applicant to verify that the as-built SB&PC system implements these actions. Accordingly, based on identified SB&PC system actions and their verification in the ITAAC, the staff finds that the requirements of GDC 10 are addressed for the SB&PC system.

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the SB&PC system. DCD Tier 2, Revision 9, Sections 7.7.5.2 and 7.7.5.6, specify the SB&PC system's manual and automatic controls and status indications. For example, the MCRP operator interface within the N-DCIS contains controls needed for SB&PC system operation. DCD Tier 2, Revision 9, Section 7.7.5.3.2, indicates conformance to GDC 13. DCD Tier 1, Revision 9, Section 2.2.9, includes ITAAC for the applicant to verify that the as-built SB&PC system implements the required automatic functions and operator controls. The staff reviewed monitoring and controls provided for the plant transient response to normal load changes and AOOs under the effects of the control system failures in Section 7.7.5.2.1 of this report and finds them acceptable. The staff concludes that the SB&PC system is capable of maintaining system variables within prescribed operating ranges and implementing contingency actions if such variables reach limiting conditions. Accordingly, based on identified SB&PC system monitoring and controls and their verification in the ITAAC, the staff finds that the requirements of GDC 13 are adequately addressed for the SB&PC system.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the SB&PC system. SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate I&C system contributions to design margin for reactor coolant systems. The staff reviewed DCD Tier 2, Revision 9, Section 7.7.5, and DCD Tier 1, Revision 9, Section 2.2.9, regarding the features of manual and automatic control described in the SB&PC system I&C that facilitate the capability to maintain plant variables within prescribed operating limits and over their anticipated ranges for normal operation, AOOs, and accident conditions, as appropriate, to assure adequate safety. DCD Tier 2, Sections 7.7.5.2 and 7.7.5.6, specify the SB&PC system's manual and automatic controls and status indications. Examples are the control signals provided to the TGCS to control the TCVs and the SB&PC system control of the TBVs. DCD Tier 1, Section 2.2.9, includes the ITAAC for the applicant to verify that the as-built SB&PC system implements the required automatic functions and operator controls. Accordingly, based on the identified SB&PC system I&C and their verification in the ITAAC, the staff finds that the requirements of GDC 15 are adequately addressed for the SB&PC system.

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the SB&PC system. In Section 7.1.1.3.6 the staff evaluated that GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should evaluate if there exists I&C available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.5.3.2 specifies that the SB&PC conforms to GDC 19. DCD Tier 2, Revision 9, Section 7.7.5.6.3, provides information on the SB&PC system for performance monitoring in the MCR through the N-DCIS. DCD Tier 2, Revision 9, Sections 7.7.5.2 and 7.7.5.6, specify the SB&PC system's manual and automatic controls and status indications. DCD Tier 1, Revision 9, Section 2.2.9, includes the ITAAC for the applicant to verify that the as-built SB&PC system implements the required automatic functions and operator controls. Based on the above, including the features for manual and automatic control described in DCD Tier 2, Revision 9, Section 7.7.5, the staff finds that the requirements of GDC 19 are adequately addressed for the SB&PC system.

DCD Tier 2, Revision 9, Section 7.7.1.1.2, identifies that the SB&PC system controls reactor pressure during plant startup, power operation, and shutdown modes. While pressure does affect reactivity, pressure is not a primary control function compared to functions controlled by the RC&IS and FWCS. Accordingly, the SB&PC system is not a reactivity control system, and the staff accepts that GDC 28 and 29 do not apply to the SB&PC system.

Therefore, the staff finds that the SB&PC system adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 above for a nonsafety system and that there is reasonable assurance that this system will be able to accomplish its designed function in a reliable manner, when built and tested according to DCD Tier 2, Revision 9, and the DCD Tier 1, Revision 9, ITAAC.

7.7.5.3 *Conclusion*

Based on the above, the staff concludes that (1) there is reasonable assurance that the SB&PC system conforms to the applicable requirements, which include GDC 1, 10, 13, 15, 19, and 24; 10 CFR 52.47(b)(1); 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h); (2) adequate high level functional requirements are identified; and (3) sufficient ITAAC are included in DCD Tier 1, Revision 9, to verify that the design is completed in compliance with the applicable requirements.

7.7.6 Neutron Monitoring System - Nonsafety Subsystems

7.7.6.1 *Summary of Technical Information*

The nonsafety portion of the NMS, (NMS(N)) has two nonsafety subsystems, the AFIP subsystem and the MRBM subsystem. The AFIP subsystem is intended as an upgraded replacement for the traversing incore probe (TIP) or automated traversing incore probe (ATIP) used in the current BWR fleet.

7.7.6.1.1 Automated Fixed Incore Probe

The purpose of the AFIP subsystem is to provide sufficient axial and radial neutron flux monitoring to support the determination of three-dimension core power distribution and to provide an automated mode of LPRM calibration by direct interface with the plant computer function of the N-DCIS.

The AFIP subsystem comprises AFIP subsystem sensors and their associated cables, as well as the signal processing electronic unit. The AFIP subsystem sensors, unlike the former TIP and ATIP, are installed permanently within the LPRM assemblies. Within each LPRM assembly in the core, seven AFIP subsystem sensors are evenly distributed axially along the LPRM assembly. Consequently, there are AFIP subsystem sensors at and between all LPRM locations. The AFIP subsystem sensor cables are routed within the LPRM assembly and then out of the RPV through the LPRM assembly penetration to the vessel. The AFIP subsystem generates signals proportional to the axial power distribution at the radial core locations of the LPRM detector assemblies. The AFIP subsystem signal range is sufficiently wide to accommodate the corresponding local power range from approximately 5 percent to 125 percent of reactor rated power.

The AFIP subsystem data collection and processing sequences are fully automated, with manual control available. The AFIP subsystem signals are used to calibrate the LPRM detectors and to determine the power distribution in the reactor core and the reactor protection parameters.

The power for the AFIP subsystem is supplied from the nonsafety instrument 120-volts ac supply power source. The power for the AFIP subsystem logic is supplied from redundant, nonsafety instrument 120-volts ac UPSs.

7.7.6.1.2 Multichannel Rod Block Monitor

The purpose of the MRBM subsystem is to monitor signals from the systems that observe the neutron flux and to provide a signal to the RC&IS to block rod movement if the MRBM subsystem signal exceeds a preset rod block setpoint to prevent fuel damage.

The MRBM subsystem logic receives input signals from the LPRMs and the APRMs of the NMS. It also receives control rod status data from the RAPI subsystem of the RC&IS to determine when rod withdrawal blocks are required. The MRBM subsystem uses the LPRM signals to detect local power change during the rod withdrawal. If the MRBM subsystem signal, which is based on averaged LPRM signals, exceeds a preset rod block setpoint, a control rod block demand is issued. The MRBM subsystem monitors the core in four-by-four fuel bundle regions where control rods are being withdrawn. The MRBM subsystem algorithm covers the monitoring of multiple regions simultaneously, depending upon the size of the gang of rods being withdrawn. Because it monitors more than one region, it is called the “multichannel rod block monitor” as compared to the original “single channel rod block monitor.” The MRBM subsystem is a dual channel system and is not a safety system.

The power supply for the MRBM subsystem is from the nonsafety 120-volts ac uninterruptible buses in two different load groups.

7.7.6.2 *Staff Evaluation*

7.7.6.2.1 Evaluation of NMS(N) Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations identified in SRP Section 7.7 and discusses the attributes that are common to the control systems described in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those major design considerations and information unique to the NMS(N).

(1) Design Basis

The NMS(N) has two nonsafety subsystems, the AFIP subsystem and the MRBM subsystem. Once the RPS senses a condition requiring actuation of the reactor protection features, neither the AFIP subsystem nor the MRBM subsystem are required to shut down the reactor or maintain it in a safe state. Thus, neither the AFIP subsystem nor the MRBM subsystem performs or ensures any safety function; therefore, the AFIP subsystem and MBRM subsystems have no safety design basis. However, these systems are important to overall safe operation of the plant. The AFIP subsystem provides axial and radial neutron flux distribution to support the determination of three-dimension core power distribution, sufficient axial neutron flux monitoring with corresponding axial position, and a totally automated mode of LPRM calibration by direct interface with the plant computer function of the N-DCIS. Other than providing input, the AFIP subsystem has no control function. The MRBM subsystem monitors control rod movement, issues a rod block signal to the RC&IS to block rod movement if the MRBM subsystem signal exceeds a preset rod block setpoint to prevent fuel damage, and provides values to the N-DCIS. Other than this rod block function, the MRBM subsystem has no other control function. The staff finds that the NMS(N) includes the necessary features to support manual and automatic control of process variables within prescribed operating limits. Accordingly, based on the above discussion and information provided in DCD Tier 2, Revision 9, Section 7.7.6, as well as the verification of these controls and functions through applicable DCD Tier 1, Revision 9, Section 2.2.5, Table 2.2.5-4, ITAAC, the staff finds that the design bases are adequately addressed for the NMS(N).

(2) Effects of Control System Operation on Accidents

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.6 and Chapter 15, regarding the effects of NMS(N) operation on accidents. The staff verified that the safety analysis considers the effects of both NMS(N) system action and inaction in assessing the transient response of the plant for accidents and AOOs. Again, the NMS(N) is a monitoring system with no direct control other than issuing rod blocks. The MRBM subsystem provides regional and axial neutron flux monitoring and issues a rod block order to the RC&IS, if a MRBM subsystem parameter limit is exceeded, as a preemptive function to prevent a reactor scram during a control rod withdrawal error. DCD Tier 2, Revision 9, Table 15.1-6, identifies that, during control rod withdrawal error, four sources could provide a rod block depending on the reactor power level: SRNM period, RWM, ATLM , and the MRBM. The control rod withdrawal error events are: (1) Control Rod Withdrawal Error during Power Operation with ATLM Failure, (2) Control Rod Withdrawal Error During Power Operation, (3) Control Rod Withdrawal Error During Startup, and (4) Control Rod Withdrawal Error During Refueling. Credit is taken because one of the rod block sources, such as the MRBM, is sufficient to mitigate any of these four control rod withdrawal errors. Even though the MRBM subsystem is credited in mitigating these four transients, the safety analysis does not depend on the MRBM subsystem for safe shutdown of the nuclear reactor since the safety function is provided by the SRNM period rod block. The AFIP subsystem data are used to assist in calculating neutron flux distribution bundle and sub-bundle powers, to calibrate the LPRM. While the AFIP subsystem may provide useful information on the neutron flux distribution in a transient investigation, there is no rod block from the AFIP subsystem or credit taken in the safety analysis of DCD Tier 2, Revision 9, Chapter 15. Inaction of the AFIP subsystem or MRBM subsystem would indicated a failed component and is covered under the effects of control system failures below. Based on the above information, a review of DCD Tier 2, Revision 9, Section 7.7.6 and Chapter 15, and the discussion under the effects of control system failures below, the staff finds that the safety analysis considers the effects of both

NMS(N) system action and inaction in assessing the transient response of the plant for accidents and AOOs.

(3) Effects of Control System Failures

The staff reviewed DCD Tier 2, Revision 9, Chapters 15 and Section 7.7.6, regarding the effects of NMS(N) failures. The staff verified that the failure of any NMS(N) system component does not cause plant conditions more severe than those described in the analysis of AOOs in DCD Tier 2, Chapter 15. The effect of a complete failure of the MRBM subsystem removes one of the four different functions that is a potential source of a rod block in case of a control rod withdrawal error. The function that would actually provide the rod block depends on reactor power. As discussed under the effects of control system operation on accidents above, DCD Tier 2, Revision 9, Table 15.1-6, notes four control rod withdrawal error events, in which any one of the rod blocks would be sufficient to mitigate the event without requiring a reactor scram. DCD Tier 2, Table 15.1-6, shows that the bounding condition for a complete failure of the MRBM subsystem during reactor startup is the transient labeled, "Control Rod Withdrawal Error During Startup With Failure of Control Rod Block," which is mitigated by the SRNM Period Scram. During power operation, the ATLM is the primary protective rod block on a control rod withdrawal error and the MRBM subsystem is the secondary source for a rod block.

DCD Tier 2, Revision 9, Table 15.1-6, also shows the control rod withdrawal error event labeled, "Control Rod Withdrawal Error during Power Operation with ATLM Failure," which is mitigated by the MRBM subsystem rod block (assuming the control rod is located above the RWM controlled region). Assuming that both the dual redundant MRBM subsystem and the dual redundant ATLM both fail at the same time is beyond design basis and the single failure criteria. If the MRBM subsystem fails, the ATLM is still available. In general, if the MRBM subsystem fails such that a rod block is not given when required, and the operator does not detect and correctly identify the problem and take appropriate action, then either an alternate rod block or the RPS would have to respond. A failed MRBM subsystem issuing an erroneous control rod block is in the conservative direction.

A massive failure of the AFIP subsystem logic that provides incorrect data is a problem worse than a failure to provide any data. Again this would not stop a scram. Testing, self-diagnostic, continued operation, data comparison, and defense in depth significantly reduce the probability of such an event. The AFIP subsystem is a passive monitoring system, and does not directly control the control rods or equipment. Also, TS criteria exist for the number of permitted failed AFIP subsystem detectors and their location. Based on the above and DCD Tier 2, Revision 9, Section 7.7.6 and Chapter 15, Tables 15.1-5 and 15.1-6, the staff finds that the occurrences and events discussed in DCD Tier 2, Chapter 15, bound any failures of the NMS(N) and, therefore, the staff finds that failures of the NMS(N) do not cause plant conditions more severe than those described in DCD Tier 2, Chapter 15. In RAI 7.7-12, the staff requested that the applicant provide analyses that evaluate the effects of control systems failures. RAI 7.7-12 was being tracked as an open item in the SER with open items. In response, the applicant stated that the Chapter 15 analysis assumed the nonsafety subsystems of the AFIP subsystem and MRBM subsystem to be operational. The applicant does not specifically analyze the individual failures of these systems since in applying the single failure criteria, if the ATLM is not available, then the MRBM subsystem would be available, and if the MRBM subsystem failed, then the ATLM would be available. However, as discussed above, the applicant does analyze a bounding failure of these systems since the applicant analyzes in DCD Tier 2, Chapter 15 a control rod withdrawal error during startup with failure of control rod block. In the unlikely failure of all rod blocks, the safety systems would act to prevent or mitigate an accident. The staff finds that the response is acceptable since DCD Tier 2, Revision 9, Chapter 15 includes events that bound the failures of the NMS(N). Based on the above and the applicant's response, RAI 7.7-12 regarding the NMS(N) is resolved.

(4) Effects of Control System Failures Caused by Accidents

The staff reviewed DCD Tier 2, Revision 9, Chapters 15 and Section 7.7.6, regarding the effects of NMS(N) failures caused by accidents. The NMS(N) consists of cabinets, or panels, that contain special electronic/electrical equipment modules for performing the NMS(N) logic in the reactor building and control building. Accidents could potentially damage AFIP subsystem cabling under the RPV. Even if all cables were destroyed, the reactor scram would not be affected. Sensors and transmitters providing input to the NMS(N) could be damaged. Fire or earthquake potentially could cause one or more functions to fail and damage the electronic equipment causing an NMS(N) function to fail. The potential for such effects is significantly reduced through D3 in NMS(N) sensors, NMS(N) environmental EQ, redundant modules in different cabinets, and a fire protection design. Based on the above information; DCD Tier 2, Revision 9, Section 7.7.6 and Chapter 15; and failures discussed under the effects of control system failures above, the staff finds that the consequential effects of AOOs and accidents do not lead to NMS(N) failures that would result in consequences more severe than those described in DCD Tier 2, Revision 9, Chapter 15. Based on the above, the staff finds that the effects of NMS(N) failures caused by accidents are adequately addressed.

(5) Potential for Inadvertent Actuation

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.6, to identify design measures that limit the potential for inadvertent actuation. Examples of such design features are discussed below.

The design has seven gamma thermometer (GT) type neutron detectors per LPRM string and LPRM strings distributed throughout the reactor core. This large number of GT detectors distributed throughout the reactor core allows a limited number of failed GT detectors before the entire AFIP is considered inoperable. The loss of a limited number of detector GTs, while undesirable, can be tolerated in support of D3. There are TS criteria for the number of GT detectors permitted failed and their location.

The MRBM subsystem is effectively a D3 feature. If the MRBM subsystem does not provide a control rod block when rod movements exceed rod movement restrictions, an alternate rod block could be generated by the SRNM fast period, RWM, ATLM, or APRM of the NMS(N) (safety portion). The MRBM subsystem logic can issue a rod block signal used in the RC&IS logic to enforce rod blocks. The logic attempts to anticipate situations from the NMS signals that are approaching, or may approach, an unsafe condition and attempts to act before the NMS signal requires a reactor scram. The rod blocks prevent fuel damage by ensuring that the MCPR and MLHGR do not violate fuel thermal safety limits. Once a rod block is initiated, manual action is required by the operator to reset the system. Furthermore, the MRBM subsystem is a dual channel system with the power supply from the nonsafety 120-volts ac uninterruptible buses in two different load groups.

Based on the above, the staff finds that the NMS(N) design limits the potential for inadvertent actuation.

7.7.6.2.2 Evaluation of NMS(N) Compliance with Regulations

Section 7.7.0.4 of this report lists and discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for the control systems identified in DCD Tier 2, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the NMS(N).

The staff reviewed DCD Tier 2, Revision 9, Section 7.7.6, and DCD Tier 1, Revision 9, Section 2.2.5, in accordance with SRP Sections 7.7 and 14.3.5, to verify that 10 CFR 52.47(b)(1) is adequately addressed for the NMS(N) as nonsafety subsystems. DCD Tier 1, Section 2.2.5, documents the NMS ITAAC requirements. While NMS(N) has no DAC, DCD Tier 1, Table 2.2.5-4, defines AFIP and MRBM subsystem as nonsafety subsystems of the NMS and verifies the MRBM subsystem main channel bypasses. Accordingly, based on information in DCD Tier 1, Revision 9, Section 2.2.5, Table 2.2.5-4, and DCD Tier 2, Revision 9, Chapter 7; information discussed herein; and identified NMS(N) I&C and their verification in the ITAAC, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed for these nonsafety subsystems.

To determine compliance with GDC 10, 13, 15, 19, 28, and 29, the staff reviewed DCD Tier 2, Revision 9, Section 7.7.6, to verify that the applicable GDC specified in SRP Section 7.7 is adequately addressed for the NMS(N) as a nonsafety system. DCD Tier 2, Revision 9, Section 7.7.3.3.2, states that the NMS(N) conforms to GDC 13, 19, 28, and 29.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the NMS(N). SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate I&C system contributions to design margin for reactor core and coolant systems. The AFIP subsystem is a fully automated measurement, data collection, data amplification, and calculation subsystem. The AFIP subsystem collects three-dimensional neutron flux readings and uses these in the calibration of the LPRMs. The subsystem is also a measurement of neutron flux and calculation subsystem to support control rod movement monitoring. Since the NMS(N) does not control any parameter directly that may affect reactor core or associated coolant system margins and the NMS(N) is addressed in GDC 13 instrumentation requirements, the staff accepts that GDC 10 does not apply to the NMS(N).

GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. The staff evaluated whether GDC 13 is adequately addressed for the NMS(N). The AFIP subsystem is a fully automated measurement, data collection, data amplification, and calculation subsystem. The AFIP subsystem collects three-dimensional neutron flux readings and uses these in the calibration of the LPRMs. Other than providing input, the AFIP subsystem has no control function. The MRBM subsystem is also a measurement of neutron flux and calculation subsystem to support control rod movement monitoring. Other than this rod block function, the MRBM subsystem has no other control function. DCD Tier 2, Revision 9, Section 7.7.6.2, specifies the automatic and manual controls of the AFIP subsystem and MRBM. DCD Tier 2, Revision 9, Section 7.7.6.3.2, indicates conformance to GDC 13. DCD Tier 1, Revision 9, Section 2.2.5, includes ITAAC for the applicant to verify that the as-built NMS(N) implements the required automatic functions and operator controls. Based on the above discussion; DCD Tier 2, Revision 9, Section 7.7.6; and verification in the ITAAC, the staff finds that GDC 13 is adequately addressed.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the NMS(N). SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate the I&C system contributions to design margin for reactor coolant systems. DCD Tier 2, Revision 9, Section 7.7.6.3.2, does not state conformance to GDC 15. The AFIP subsystem is a fully automated measurement, data collection, data amplification, and calculation subsystem. The AFIP subsystem collects three-dimensional neutron flux readings and uses these in the calibration of the LPRMs. The MRBM subsystem is also a measurement of neutron flux and calculation subsystem to support control rod movement monitoring. Since the NMS(N) does not control any parameter that may affect RCPB margins, the staff finds that GDC 15 does not apply to the NMS(N).

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the NMS(N). In Section 7.1.1.3.6 the staff finds that GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should evaluate if there exists I&C available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.6.3.2, specifies that the NMS(n) conforms to GDC 19. The AFIP subsystem automatically collects important data and passes them on to the MCR and the Process Computer Function. The AFIP subsystem is a fully automated measurement, data collection, data amplification, and calculation subsystem. The AFIP subsystem collects three-dimensional neutron flux readings and uses these in the calibration of the LPRMs. Other than providing input, the AFIP subsystem has no control function. The MRBM subsystem is also a measurement of neutron flux and calculation subsystem to support control rod movement monitoring. Other than this rod block function, the MRBM subsystem has no other control function. There are bypass controls in the MCR for the MRBM. DCD Tier 2, Revision 9, Section 7.7.6.2, specifies the automatic and manual controls of the AFIP subsystem and the MRBM. DCD Tier 1, Revision 9, Section 2.2.5, includes the ITAAC for the applicant to verify that the AFIP and MRBM subsystem are implemented as nonsafety subsystems in the NMS. Based on the above discussion; DCD Tier 2, Revision 9, Section 7.7.6; and verification by DCD Tier 1, Revision 9, Section 2.2.5 ITAAC, the staff finds that the requirements of GDC 19 are adequately addressed.

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. The staff evaluated whether GDC 28 is adequately addressed for the NMS(N). SRP Appendix 7.1-A states that GDC 28 imposes functional requirements on I&C interlock and control systems to the extent they are provided to limit reactivity increases to prevent or limit the effect of reactivity accidents. DCD Tier 2, Revision 9, Section 7.7.6.3.2, specifies that the NMS(N) conforms to GDC 28. The MRBM subsystem provides a direct protective control function through the MRBM subsystem rod block when rod movement restrictions are violated. The AFIP subsystem is a monitoring and measurement system essential for calibrating the LPRM and providing the data for determining the core power distribution and thus the RPS protective parameters. The MRBM subsystem is an anticipatory defense system to protect the reactor core. DCD Tier 2, Revision 9, Chapter 15 and Tables 15.1-5 and 15.1-6, identify actuations and other actions that reduce the need for the actuation of protection and safety systems to mitigate AOOs. DCD Tier 2, Revision 9, Section 7.7.6, includes functions of the NMS(N) that significantly contribute to the design bases of the NMS(N) to maintain the reactor core and the reactivity limits within appropriate margins and to mitigate AOOs. An example is the rod block protective function of the MRBM. DCD Tier 1, Revision 9, Section 2.2.5, includes ITAAC to verify that the AFIP and MRBM subsystem are implemented as nonsafety subsystems in the NMS. Accordingly, based on the identified NMS(N) actions and their verification in the ITAAC, the staff finds that the requirements of GDC 28 are adequately addressed for the NMS(N).

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed for NMS(N). SRP Appendix 7.1-A, identifies that GDC 29 is addressed by conformance, as applicable, to GDC 20-25 and GDC 28. Since the NMS(N) is a reactivity control system and not a protection system, GDC 20-23 and 25, which are requirements for protection systems, are not applicable to the NMS(N), Accordingly, GDC 29 is addressed by conformance as applicable to GDC 24 and 28. Section 7.7.0.4 of this report evaluates conformance of N-DCIS control systems, including the NMS(N), to GDC 24. Conformance of the NMS(N) to GDC 28 is evaluated above. DCD Tier 2, Revision 9, Section 7.7.6.3.2, specifies that the NMS(N) conforms to GDC 29. In addition, DCD Tier 1, Revision 9, Section 2.2.5, includes ITAAC to verify that the AFIP and MRBM subsystem are implemented as nonsafety subsystems in the NMS. Based on the information above and adequate consideration of the requirements of GDC 24 and 28, the staff finds that the requirements of GDC 29 are adequately addressed for the NMS(N).

Therefore, the staff finds the NMS(N) adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 above for a nonsafety system and that there is reasonable assurance that this system will be able to accomplish its designed function in a reliable manner when built and tested according to DCD Tier 2, Revision 9 and the DCD Tier 1, Revision 9, ITAAC.

7.7.6.3 *Conclusion*

Based on the above, the staff concludes that (1) there is reasonable assurance that the NMS(N) made up of the AFIP subsystem and the MRBM subsystem, conforms to the applicable requirements, which include GDC 1, 13, 19, 24, 28, and 29; 10 CFR 52.47(b)(1); 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h); (2) adequate high level functional requirements are identified, and (3) sufficient ITAAC are included in DCD Tier 1, Revision 9, to verify that the design is completed in compliance with the applicable requirements.

7.7.7 Containment Inerting System

7.7.7.1 *Summary of Technical Information*

The objective of the CIS is to preclude combustion of hydrogen and prevent damage to essential equipment and structures by providing an inerted containment environment. The CIS is designed to establish an inert atmosphere (i.e., less than 4 percent oxygen by volume) throughout the containment in less than 4 hours and less than 2 percent oxygen by volume in the next 8 hours following an outage. The CIS is also designed to maintain the containment oxygen concentration below the maximum permissible limit (4 percent) during normal power operation to ensure an inert atmosphere and to minimize hydrogen burn inside the containment in case of an event that would release hydrogen. The CIS is capable of reaching a volumetric oxygen concentration of greater than or equal to 19 percent within 12 hours after de-inerting begins. DCD Tier 2, Revision 9, Sections 7.7.7 and 6.2.5.2, provide further details. DCD Tier 2, Revision 9, Figure 6.2-29, depicts a simplified CIS system. Further, the CIS is designed to maintain a positive pressure in the primary containment during normal, abnormal, and accident conditions.

The CIS I&C is provided to monitor the process variables and operate the system processes during startup, normal, and abnormal reactor operation. The CIS is operated from the MCR.

7.7.7.2 *Staff Evaluation*

7.7.7.2.1 Evaluation of Containment Inerting System Conformance with Acceptance Criteria

Section 7.7.0.3 of this report lists the major design considerations identified in SRP Section 7.7 and discusses the attributes that are common to the control systems described in DCD Tier 2, Revision 9, Section 7.7. This section will discuss and evaluate only those major design considerations and information that are unique to the CIS.

(1) Design Basis

The CIS is not a safety system except for the containment isolation function which is outside the scope of this evaluation. The CIS is not required for safe shutdown of the plant. Therefore, the CIS has no safety design basis. Failure of the nonsafety I&C components does not adversely affect any safety function. The CIS does have a power generation (nonsafety) design basis. The nonsafety design basis of the CIS is to establish and maintain an inert atmosphere in the containment, to maintain a positive pressure in the containment, and to perform continuous leakage rate monitoring. The staff finds that the CIS includes the necessary features for manual and automatic control of process variables within prescribed operating limits.

(2) Effects of Control System Operation on Accidents

The staff reviewed DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7 and Chapter 15, regarding the effects of CIS operation on accidents. During AOOs and accident conditions, the containment inerting function is passive and does not require the operation of the CIS. The CIS can be used under post accident conditions for containment atmosphere dilution to maintain inert conditions. CIS inaction indicates a failed component and is discussed below. Based on the above information and a review of DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7 and Chapter 15, the staff finds the effect of the CIS on accidents is adequately addressed.

(3) Effects of Control System Failures

The staff reviewed DCD Tier 2, Revision 9, Sections 6.2.5.2, Section 7.7.7 and Chapter 15, regarding the effects of CIS failures. Even if CIS functions were to fail, a reactor shut down to a safe condition would not be prevented. During AOOs and accident conditions, the containment inerting function is passive and does not require the operation of the CIS. In the case of certain failures, significant portions of the CIS equipment can be operated locally and manually as a backup, as well as from the MCR. Based on the above, a review of DCD Tier 2, Sections 6.2.5.2 and 7.7.7 and Chapter 15, the staff finds the effects of the CIS on accidents are adequately addressed.

(4) Effects of Control System Failures Caused by Accidents

The staff reviewed DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7 and Chapter 15, regarding the effects of CIS failures caused by accidents. Accidents could potentially damage CIS I&C cabling, controls in cabinets, monitors, sensors, and transmitters. Fire or earthquake potentially could damage electronic equipment causing one or more CIS functions to fail. Even if CIS functions were to fail, a reactor scram would not be prevented. The potential for such effects is significantly reduced through environmental and a fire protection design. The CIS instrument lines penetrating containment comply with the guidance of RG 1.151. Based on the above information; a review of DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7 and Chapter 15; and the failures discussed above, the staff finds that the consequential effects of AOOs and accidents do not lead to CIS failures that would result in consequences more severe than those described in DCD Tier 2, Chapter 15. Based on the above, the staff finds that the effects of CIS failures caused by accidents are adequately addressed.

(5) Potential for Inadvertent Actuation

The staff reviewed DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7, to identify design measures that limit the potential for inadvertent actuation. Examples of such design features are described below.

In the case of certain failures, significant portions of the CIS equipment can be operated locally as a backup as well as from the MCR. Operator training and operating procedures are the expected method to reduce inadvertent actuation. The CIS cannot adversely affect the safety systems. Unlike many of the other control systems discussed in DCD Tier 2, Revision 9, Section 7.7, an inadvertent actuation of the CIS is not as serious as it would be for many other systems. This system is used to meet technical specifications requirements for some of the parameters such as the maximum permissible limit (4 percent) on oxygen in the containment during normal power operation. The staff finds that inadvertent actuation is adequately addressed for the CIS.

7.7.7.2.2 Evaluation of Containment Inerting System Compliance with Regulations

Section 7.7.0.4 of this report lists and discusses the common design attributes and methods for complying with the regulations required by SRP Section 7.7 for control systems identified in DCD Tier 2, Section 7.7. This section will discuss and evaluate only those regulations with unique methods of compliance for the CIS.

The staff reviewed DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7, and DCD Tier 1, Revision 9, Section 2.15.5, in accordance with SRP Sections 7.7 and 14.3.5, to verify that 10 CFR 52.47(b)(1) is adequately addressed for the CIS. DCD Tier 1, Revision 9, Section 2.15.5, documents the CIS ITAAC requirements, which include verifying that the containment can be inerted to less than or equal to 4 percent oxygen by volume and that the drywell temperature indications are retrievable in the main control room. The staff finds that additional ITAAC on control functions are not needed since the CIS has no safety functions, and it is not required by regulation. Based on information reviewed in DCD Tier 1, Revision 9; DCD Tier 2, Revision 9, Chapters 6 and 7; information discussed herein; and identified CIS I&C and their verification in the ITAAC, the staff finds that the requirements of 10 CFR 52.47(b)(1) are adequately addressed.

To determine compliance with GDC 10, 13, 15, 19, 28, and 29, the staff reviewed DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7, to verify that the applicable GDC specified in SRP Section 7.7 are adequately addressed for the CIS as a nonsafety system. DCD Tier 2, Section 7.7.7.3.2, states that the CIS conforms to GDC 13 and 19.

GDC 10 requires that the reactor core, associated coolant, control, and protection systems be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The staff evaluated whether GDC 10 is adequately addressed for the CIS. SRP Appendix 7.1-A for GDC 10 states that the staff review should evaluate the I&C system contributions to design margin for reactor core and coolant systems. The CIS is a system used to assure that the containment atmosphere is inerted with nitrogen gas to displace the oxygen and remove the possibility of a hydrogen/oxygen explosion. The CIS does not have any functions associated with reactor reactivity control, the reactor coolant boundary, or cooling water and associated AOOs. GDC 10 is associated with reactor design and does not apply to the CIS.

DCD Tier 2, Revision 9, Section 7.7.7.3.2, states that the CIS conforms to GDC 13. The CIS operation is manually or automatically activated from the MCR by aligning corresponding valves through remote manual control switches. The CIS has three control modes: inerting, makeup, and de-inerting. DCD Tier 2, Revision 9, Section 7.7.7.5.1, specifies the automatic and manual controls for inerting and de-inerting the containment. DCD Tier 2, Section 7.7.7.5.1, also specifies the automatic control for the makeup mode, which maintains the containment pressure. DCD Tier 2, Revision 9, Section 7.7.7.5.3, specifies the alarms and status indications in the MCR to support CIS operations. During AOOs and accident conditions, the containment inerting function is passive and does not require the operation of the CIS. DCD Tier 2, Section 7.7.7.3.2, indicates conformance to GDC 13. Based on the above discussion and a review of DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7, the staff finds that the requirements for GDC 13 are adequately addressed for CIS.

GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The staff evaluated whether GDC 15 is adequately addressed for the CIS. SRP Appendix 7.1-A for GDC 15 states that the staff review should evaluate the I&C system contributions to design margin for reactor coolant systems. GDC 15 is associated with systems having an influence on the reactor coolant system design and does not apply to the CIS.

GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. The staff evaluated whether GDC 19 is adequately addressed for the CIS. In Section 7.1.1.3.6 the staff finds that GDC 19 is adequately addressed with the exception of the operation of specific I&C systems. SRP Appendix 7.1-A states that the review should evaluate if there exists I&C available to operate the nuclear power unit under normal and accident conditions. DCD Tier 2, Revision 9, Section 7.7.7.3.2, specifies that the CIS conforms to GDC 19. DCD Tier 2, Revision 9, Section 7.7.7.5.1, describes the CIS logic, interlocks, and general I&C. The CIS operation is manually or automatically activated from the MCR by aligning corresponding valves through remote manual control switches. The CIS has three control modes: inerting, makeup, and de-inerting. DCD Tier 2, Section 7.7.7.5.1, specifies logic and interlocks for the automatic and manual controls for inerting and de-inerting the containment. DCD Tier 2, Section 7.7.7.5.1, also specifies the automatic control for the makeup mode, which maintains the containment pressure. DCD Tier 2, Revision 9, Section 7.7.7.5.2, specifies I&C including drywell pressure sensors, containment temperature and humidity sensors, flow metering devices, oxygen analyzers, and interfaces with other systems. DCD Tier 2, Revision 9, Section 7.7.7.5.3, specifies the alarms and status indications in the MCR to support CIS operations. During AOOs and accident conditions, the containment inerting function is passive and does not require the operation of the CIS. Based on the above discussion and a review of DCD Tier 2, Revision 9, Sections 6.2.5.2 and 7.7.7, the staff finds that the requirements for GDC 19 are adequately addressed for the CIS.

GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. The staff evaluated whether GDC 28 is adequately addressed for the CIS. GDC 28 is associated with systems having an influence on reactivity control and does not apply to the CIS.

GDC 29 requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs. The staff evaluated whether GDC 29 is adequately addressed for the CIS. GDC 29 is associated with systems having an influence on the RPS and reactivity control and does not apply to CIS.

Therefore, the staff finds that the CIS adequately addresses the relevant regulatory criteria listed in Section 7.7.0.1 above for a nonsafety system and that there is reasonable assurance this system will be able to accomplish its designed function in a reliable manner, when built and tested according to DCD Tier 2, Revision 9 and the DCD Tier 1, Revision 9, ITAAC.

7.7.7.3 *Conclusion*

Based on the above, the staff concludes that (1) there is reasonable assurance that the CIS conforms to the applicable requirements, which include GDC 1, 13, 19, and 24, 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h); (2) adequate high level functional requirements are identified, and (3) sufficient ITAAC are included in DCD Tier 1, Revision 9, to verify that the design is completed in compliance with applicable requirements.

7.7.8 Conclusion on Control System

The staff concludes that design of the control systems identified in DCD Tier 2, Revision 9, Section 7.7, which are evaluated in this section, is acceptable and meets the relevant requirements of GDC 1, 10, 13, 15, 19, 24, 28, and 29; 10 CFR 50.34(f); 10 CFR 50.55a(a)(1); and 10 CFR 50.55a(h). This conclusion is based on the review of the design of these nonsafety control systems, as described in DCD Tier 1, Revision 9 and DCD Tier 2, Revision 9, and the staff's finding that these control systems are appropriately designed and are of sufficient quality to 1) minimize the potential for challenges to the safety systems; 2) provide manual and automatic control features capable of maintaining system variables within prescribed operating ranges; 3) protect the instrumentation from threats within the environment such as freezing in sensing lines; 4) protect the instrumentation from natural phenomenon; 5) provide appropriate isolation from safety systems; 6) provide enhanced reliability through redundancy, diversity, defense against potential CCFs, and minimizing the probability of inadvertent actuation; and 7) provide I&C such that plant safety does not depend on the response or lack of response or failure of these control systems, but can supplement and support the features of the safety systems.

7.8 <u>Diverse Instrumentation and Control Systems</u>

7.8.1 Regulatory Criteria

The objectives of the review of DCD Tier 1, Revision 9, Section 2.2 and DCD Tier 2, Revision 9, Section 7.8, are to ensure that the ATWS mitigation systems and equipment are designed and installed in accordance with the requirements of 10 CFR 50.62 and that other diverse I&C systems within the scope of this section comply with the staff acceptance criteria on D3 and the design of the diverse I&C systems, including the ATWS mitigation system, the DPS, and the common mode failure defenses, conform to SRP guidance.

Acceptance criteria in SRP Section 7.8, Revision 5, for the diverse I&C systems are based on meeting the relevant requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1, 13, 19, and 24; 10 CFR 52.47(b)(1); and 10 CFR 50.62. The acceptance criteria are also based on conforming to the guidelines of IEEE Std 7-4.3.2, as endorsed by RG 1.152, and the SRM to SECY-93-087.

7.8.2 Summary of Technical Information

The ATWS mitigation system and the DPS comprise the diverse I&C systems that are part of the D3 strategy. They provide diverse backup to the RPS and the SSLC/ESF. The ATWS mitigating logic is designed to meet the diverse shutdown requirements of 10 CFR 50.62. The ATWS mitigating logic system is implemented with the Q-DCIS (ATWS/SLC) and the N-DCIS. The nonsafety DPS (which is part of the N-DCIS) processes the nonsafety portions of the ATWS mitigation logic. It is designed to mitigate the possibility of digital protection system CCFs discussed in the SRM to SECY-93-087, Item II.Q.

The ATWS/SLC mitigation logic provides a diverse means of emergency shutdown using the SLC system for soluble boron injection. ARI, which hydraulically scrams the plant using the three sets of ARI valves of the CRD system, is also used for ATWS mitigation. The DPS implements this logic.

The DPS is a nonsafety, triple redundant system powered by redundant nonsafety load group power sources. The DPS provides diverse reactor protection using diverse set of scram logics from the RPS. The DPS provides diverse emergency core cooling by independently actuating the ECCS. The DPS performs selected containment isolation functions as part of the diverse ESF function. The scope of the DPS functions is based on the D3 strategy outlined in NEDO-33251.

The following provide mitigation of CCFs:

- Manual scram and MSIV isolation by the operator in the MCR in response to diverse parameter indications
- Availability of diverse manual initiation of the passive ECCS functions, including GDCS squib valve initiation, SRV initiation, DPV initiation, ICS initiation, ICS Vent Function, and SLC system squib valve initiation (manual initiation functions are available in the safety systems and in the DPS)
- Core makeup water capability from the condensate and feedwater system, CRD system, and FAPCS in the LPCI mode

- Long-term shutdown capability in the two redundant RSS panels, which are equipped with Division 1 and 2 controls for manual scram and MSIV closure; Division 1 and 2 safety VDUs; and a nonsafety VDU to allow monitoring and control of all plant systems (local displays of process variables in the RSS system are continuously powered and are available for monitoring at any time)

- Diverse scram, which differs from the safety RPS, using diverse hardware and software

- Diverse ESF initiation logic, which differs from the SSLC/ESF, using diverse hardware and software

- ATWS mitigation using liquid boron injection for emergency plant shutdown through the SLC system

- ATWS mitigation using ARI to hydraulically scram the plant using the three sets of ARI valves of the CRD system

- SCRRI command to the RC&IS

- SRI to hydraulically insert selected control rods with every SCRRI action

- Manual initiation capability of the ATWS mitigation functions (ARI/SLC/feedwater runback)

The following scram signals are selected for inclusion in the DPS:

- High RPV dome pressure
- High RPV water level (Level 8)
- Low RPV water level (Level 3)
- High drywell pressure
- High suppression pool temperature
- Closure of the MSIVs
- RPS scram
- SCRRI/SRI command with power levels remaining elevated

Major diverse ESF functions include the following:

- The DPS includes the ESF functions of the GDCS squib valves, SLC system squib valves, ICS, and ADS (SRVs and DPVs). The initiating logic is based on low RPV water level (Level 1).

- The DPS does not provide automatic initiation of the suppression pool equalizing function of the GDCS because it is not required for approximately 30 minutes. Therefore, manual suppression pool equalization capability is provided.

- The DPS also provides the ability to generate diverse manual ECCS actuation from the DPS displays. Manual controls are provided for ADS and GDCS injection sequenced initiation. The DPS does not provide automatic ADS and GDCS injection start on sustained high drywell pressure since this function is not required for 60 minutes.

- For the SRV or DPV opening function, three of the four solenoids on each SRV or DPV are powered by three of the four divisional safety power sources in the ESF ADS. A fourth solenoid on each SRV or DPV is powered by the nonsafety load group, with the trip logic controlled by the DPS.

- The ICS logic is configured to allow the availability of each ICS loop flow path from the four safety divisions and the DPS.

The following signals inhibit automatic initiation of the ADS by the DPS:

- Coincident low RPV water level (Level 2) and SRNM ATWS permissive signals (i.e., an SRNM signal from the NMS that is above a specified setpoint)

- Coincident high RPV pressure and SRNM ATWS permissive signals that persist for 60 seconds

The ADS inhibit logic also inhibits the ADS and GDCS injection sequenced initiation from occurring via the DPS logic. The DPS-ADS inhibit logic is also used to inhibit the DPS feedwater isolation upon high-high drywell pressure. MCR controls are provided for the above inhibit logic within the DPS under ATWS conditions.

The DPS also provides the following major isolations using 2/4 sensor logic and 2/3 processing logic. The isolation functions performed as part of the diverse ESF are “energize to actuate”:

- The MSIVs are closed upon detection of high steam flow rate, low RPV pressure, or low RPV water level (Level 2). The isolation function is performed by contacts in the 120-volts ac MSIV solenoid return circuit. The logic is enabled when the reactor is in run mode.

- The RWCU/SDC isolation valves are closed upon high differential flow rate.

- Isolation of the feedwater lines occurs upon a feedwater line break inside containment or LOCA conditions that pose a challenge to containment design pressure. The line break is sensed by differential pressure between feedwater lines coincident with high drywell pressure. A feedwater isolation also occurs upon high-high drywell pressure or high drywell pressure coincident with high drywell water level. The DPS trips the feedwater pump adjustable speed drive motor circuit breakers and closes the feedwater containment isolation valve.

- Isolation of CRD high pressure makeup water injection occurs upon on high drywell pressure coincident with high drywell level, or low level in 2/3 GDCS pools.

The DPS performs the following additional functions:

- With logic similar to the SSLC/ESF, the DPS initiates the ICS upon high RPV dome pressure, low RPV water level (Level 2), or MSIV closure to provide core cooling.

- With logic similar to the SSLC/ESF, the DPS opens the ICS lower header vent valves after 6 hours of ICS initiation.

- The DPS trips the feedwater pumps upon high RPV water level (Level 9).

- The DPS opens pool cross-connect valves between the equipment storage pool and the IC/PCCS expansion pools when a low level condition is detected in either of the IC/PCCS inner expansion pools. The DPS uses the four nonsafety level sensors in each IC/PCCS inner expansion pool which are part of FAPCS.

All safety systems have displays and controls located in the MCR that provide manual system-level actuation of their safety functions and monitoring of parameters that support those safety functions.

In addition to the manual controls and displays for the safety reactor protection and SSLC/ESF functions, the DPS also has displays and manual control functions that are independent and diverse from those of the safety protection and SSLC/ESF functions. They are not subject to the same CCF as the safety protection system components. The manual controls permit manual initiation of the SRV, DPV, GDCS, and SLC system valves, as well as the ICS. The operator is provided with a set of diverse displays separate from those supplied through the safety software platform. The displays that provide independent confirmation of the status of major process parameters include the following:

- Reactor pressure
- Reactor pressure high alarm
- RPV water level
- RPV water level high alarm
- RPV water level low alarm
- Drywell pressure
- Drywell pressure high alarm
- Drywell water level
- Drywell water level high alarm
- Suppression pool temperature
- Suppression pool temperature high alarm
- SRV solenoid-controlled valves opening
- DPV squib-initiation valves opening
- GDCS squib-initiation valves opening
- GDCS pool level
- GDCS pool level low alarm
- SLC system squib injection valves opening
- ICS operation

In addition to the controls provided by the primary safety systems, the RSS provides manual control of shutdown cooling functions and continuous local display of monitored process parameters.

7.8.3 Staff Evaluation

7.8.3.1 *Evaluation of Diverse I&C Systems Compliance with Acceptance Criteria - Major Design Considerations*

In accordance with SRP Section 7.8, the staff evaluated the following major design considerations in the review of the DPS:

(1) Design Basis

The staff evaluated the design bases described in the DCD for the DPS. The staff evaluated whether the design bases addressed the specific design requirements identified in 10 CFR 50.62, as applicable. Section 7.8.3.2 and Section 7.1.1.3.4, Item (10), of this report present the staff's evaluation of the design requirements, which finds that 10 CFR 50.62 is adequately addressed. DCD Tier 2, Revision 9, Section 7.8.1, provides a system description for the systems comprising the ATWS mitigation system and the DPS. The DPS provides backup to the RPS and the SSLC/ESF. The ATWS mitigating logic is designed to meet the diverse shutdown requirements of 10 CFR 50.62. The ATWS mitigating logic system is implemented with the ATWS/SLC and the N-DCIS.

The staff evaluated whether the design bases identify the conditions that require protective action by the DPS. As identified in DCD Tier 2, Revision 9, Section 9.3, the failure of control rods to insert in response to a valid trip demand is assumed. ATWS mitigation logic is provided to initiate the ATWS/SLC system if an SRNM power permissive exists. A delay provides sufficient time for completion of the other ATWS mitigation functions of ARI and FMCRD motor-driven run-in to shutdown the reactor. Additionally, DCD Tier 2, Revision 9, Section 7.8.1, identifies the DPS reactor trip functions that provide a diverse means of reactor shut down. A subset of the RPS scram signals is selected for inclusion in the DPS scope, which provides acceptable diverse protection results. This set of diverse protection logics for reactor scram, combined with the ATWS mitigation features, other diverse backup scram protection, and diverse ESF functions, meets BTP HICB-19.

The staff evaluated whether the design bases identify the bounding events and the bases in the analyses that are presented or referenced in DCD Tier 2, Revision 9, Chapter 15. The ATWS/SLC mitigation logic provides a diverse means of emergency shutdown using the SLC system for soluble boron injection. The shutdown functional performance requirements of the SLC system are bounded by the ATWS event performance requirements. DCD Tier 2, Revision 9, Table 15-5, provides the ATWS performance requirements and characteristics. ARI, which hydraulically scrams the plant using the three sets of air header dump valves of the CRD system, is also used for ATWS mitigation. DCD Tier 2, Revision 9, Section 7.7.2, details the ARI functionality. DCD Tier 2, Revision 9, Section 7.8.4, specifies the testing and inspection requirements. Periodic testing is performed on the ATWS/SLC and the DPS logics to verify proper operation of the DPS. DCD Tier 1, Revision 9, ITAAC Table 2.2.14-4, details validation and testing for the DPS. The staff finds the design basis consideration for the DPS to be adequately addressed. The staff also finds that the design basis consideration related to 10 CFR 50.62 to be adequately addressed, as described in Section 7.8.3.2 of this report.

(2) Quality of Components and Modules

The staff evaluated whether the quality of the DPS components and modules conforms to Generic Letter (GL) 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related." DCD Tier 1, Revision 9, Sections 2.2.14 and 3.8, adequately address the EQ of the DPS safety components. However, the EQ of the nonsafety components (ATWS/ARI) in the DPS were not addressed in DCD Tier 2, Revision 5 according to the acceptable guidance for the quality assurance of the DPS, which is included in GL 85-06. DCD Tier 2, Revision 5, Section 7.8, described the nonsafety ATWS/ARI mitigation system and the DPS as part of the diverse I&C systems. SRP Section 7.8 states that GL 85-06 provides acceptable guidance for the quality assurance of the DPS components. Additionally, SRP Section 7.8 states that the applicant should identify the test, maintenance, surveillance, and calibration procedures consistent with the guidance of GL 85-06. DCD Tier 2, Revision 5, Table 1.9-7, did not identify any differences with SRP Section 7.8. However, the DCD does not incorporate the guidance for the DPS quality, system testing, and surveillance provided in GL 85-06. While DCD Tier 2, Revision 5, Table 3.2-1, included notes for Component C-12, Item 10, and Component C-41, Item 7, stating, "A quality assurance program that meets or exceeds the guidance of GL 85-06 is applied to all Nonsafety ATWS equipment," no comparable note is provided for component C-72, the DPS. In RAI 7.8-8, the staff requested the applicant to identify in the DCD how it plans to address the EQ, quality assurance, and procedure guidance of GL 85-06 for the DPS. RAI 7.8-8 was being tracked as an open item in the SER with open items. In its response, the applicant revised DCD Tier 2, Table 3.11-1, to add systems and components associated with the DPS and to identify how the EQ program will be applied. In DCD Tier 2, Section 7.8.3, the applicant specified that the guidance contained in GL 85-06 is applied to the DPS, which includes designing and developing software used in the DPS accordance with the requirements of NEDE-33226P and assuring the quality of said software in accordance with the requirements of NEDE-33245P. The staff finds that the response is acceptable since the applicant ensured that Revision 6 of the DCD properly addresses the quality assurance of the DPS in accordance with GL 85-06. Based on the applicant's response, RAI 7.8-8 is resolved. In DCD Tier 2, Revision 6, the applicant added notes to Component C-72, the DPS, on Table 3.2-1. The notes specified the requirements to comply with GL 85-06. The staff finds the clarification acceptable.

(3) System Testing and Surveillance

The staff evaluated whether the applicant identified test, maintenance, surveillance, and calibration procedures consistent with the guidance of GL 85-06. GL 85-06 states that measures are to be established to test, as appropriate, nonsafety-related-ATWS equipment prior to installation and periodically. The staff also evaluated whether the ATWS mitigation system should be testable at power (up to, but not necessarily including, the final actuation device). DCD Tier 2, Revision 9, Section 7.8.4, specifies the DPS testing and inspection requirements. NEDE-33226P and NEDE-33245P identifies the factory tests that will be performed on I&C systems prior to installation, including the DPS. The technical specifications in DCD Tier 2, Revision 9, Chapter 16 identify the periodic tests that are performed on the DPS logics for diverse actuation of safety systems to verify proper operation of the DPS, including channel checks, channel functional tests, channel calibrations, and logic system functional tests. The Availability Controls Manual in DCD Tier 2, Revision 9, Chapter 19, Appendix A identifies the periodic tests that are performed on the ATWS/SLC and the DPS logics for diverse actuation of nonsafety systems to verify proper operation of the DPS, including channel checks, channel functional tests, channel calibrations, and logic system functional tests. The staff finds that these periodic tests conform to the guidance of GL85-06 since typical I&C tests and frequencies are identified for the DPS. DCD Tier 2, Revision 9, Sections 7.8.3.4 and 7.8.3.5, state that the design conforms to RGs 1.22 and 1.118 and BTP HICB-17 for self-test and surveillance test. The staff finds the system testing and surveillance consideration for the DPS to be adequately addressed.

(4) Use of Digital Systems

The staff evaluated whether IEEE Std 7-4.3.2, as endorsed by RG 1.152, is adequately addressed for the DPS. SRP Appendix 7.1-D provides guidance on the implementation of IEEE Std 7-4.3.2 concerning the use of digital systems. In Section 7.1.1.3.10 of this report, the staff evaluated in parallel IEEE Std 7-4.3.2 and IEEE Std 603 using the guidance in SRP Appendix 7.1-D. In RAI 7.1-99, Item D, the staff requested that the applicant include in the DAC/ITAAC the IEEE Std 7-4.3.2 criteria not already covered by DAC/ITAAC in DCD Tier 1, Sections 2.2.15 or 3.2. RAI 7.1-99 was being tracked as an open item in the SER with open items. In its response, the applicant modified DCD Tier 2, Section 7.1.6.6.1, to describe how IEEE Std 7-4.3.2 criteria are addressed by the IEEE Std 603 criteria or by NEDE-33226P and NEDE-33245P. The applicant clarified DCD Tier 1, Section 2.2.15, to state that when the IEEE Std 603 design criteria are applied to platforms relying on the use of software to perform their safety-related functions, additional criteria from IEEE Std 7-4.3.2, which augments the IEEE Std 603 criteria, also apply to the software projects as described under the applicable IEEE Std 603 criterion. In addition, the applicant rewrote the DCD Tier 1, Section 3.2, ITAAC for software in DCD Revision 9 to refer to individual components of NEDE-33226P and NEDE-33245P. Since NEDE-33226P and NEDE-33245P include applicable IEEE Std 7-4.3.2 requirements, the IEEE Std 7-4.3.2 requirements are addressed by the Revision 9, DCD Tier 1, Section 3.2, ITAAC. The staff finds that the response is acceptable since Revision 9 of the DCD describes how IEEE Std 7-4.3.2 criteria are addressed by the IEEE Std 603 criteria or by NEDE-33226P and NEDE-33245P, and the IEEE Std 7-4.3.2 criteria are included in the revised DAC/ITAAC in DCD Tier 1, Revision 9, Sections 2.2.15 or 3.2. Based on the applicant’s response and DCD changes, RAI 7.1-99, Item D is resolved.

In DCD Tier 2, Revision 8, Section 7.8.3.4, the applicant committed to comply with RG 1.152, which endorses IEEE Std 7-4.3.2. In DCD Tier 1, Revision 9, Section 2.2.14, the applicant documented detailed ITAAC requirements for the DPS. The staff finds that the concerns regarding the use of digital systems are adequately addressed.

NEDE-33226P describes the software development activities and NEDE-33245P describes software QA activities. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC to confirm that the completion of these activities and products conforms to the processes described in NEDE-33226P and NEDE-33245P and the guidelines of BTP HICB-14. Section 7.1.2 of this report provides the staff’s evaluation of software development and software QA activities.

(5) Power Supply Availability

The staff evaluated whether power sources will be available during and following a loss of offsite power. DCD Tier 2, Revision 9, Section 15.2.5.2 and Table 15.2-21, provide details of a scenario involving loss of nonemergency ac power to station auxiliaries (AOO). Loss of power generation buses produces a reactor trip (scram) signal. The event assumes normal function of the I&C and RPS. NEDO-33251, Appendix A, provides further analysis that assumes that the RPS fails to process trip signals. In this scenario the DPS, which is available and powered from dc (battery) buses, still functions to shut down the reactor at RPV Level 3. The staff finds that the DPS functionality demonstrates the versatility of the DPS in mitigating an ac power supply anomaly. Accordingly, the staff finds the power supply availability consideration for the DPS to be adequately addressed.

(6) Environmental Qualification

The staff evaluated whether the DPS equipment is qualified for the environment that could exist during the events for which the equipment is assumed to respond. The staff reviewed DCD Tier 2, Revision 9, Section 7.8.3, which stated that the guidance contained in the SRM to SECY-93-087, Item II.Q; BTP HICB-19; and GL 85-06 is applied to all diverse I&C systems and components described in this section. As discussed in Section 7.8.3.1, Item (2) of this report, the staff finds that the EQ of the DPS is adequately addressed. In DCD Tier 2, Revision 9, Section 7.8.3.4, the applicant incorporated the guidance of RG 1.89 and RG 1.209 into the harsh and mild environmental qualification programs, respectively. Section 3.11 of this report evaluates the harsh and mild EQ programs. DCD Tier 2, Revision 9, Table 3.11-1, identifies how DPS equipment is qualified. DCD Tier 2, Revision 9, Section 7.8.3.4, identifies that conformance of the DPS equipment to RG 1.89 and RG 1.209 is through the qualification programs identified in DCD Tier 2, Table 3.11-1. Based on the above, the staff finds that the EQ concern is adequately addressed for the DPS.

(7) System Status

The staff evaluated whether information related to the operation of the DPS is available in the MCR. DCD Tier 2, Revision 9, Section 7.8.1.3, describes the manual controls and displays in the MCR for the systems operated by the DPS. These are in addition to safety-related controls and displays for the safety-related systems that are initiated by the DPS. The controls and displays for the safety-related systems comply with the applicable requirements of 10 CFR 50.34(f)(2)(v)(I.D.3) and the guidelines of RG 1.47, as discussed in Section 7.1.1.3.4 of this report. The N-DCIS includes controls and displays for nonsafety-related systems initiated by the DPS.

Based on the information in DCD Tier 2, Revision 9, Section 7.8, and the information discussed herein, the staff finds that the system status indication and display are adequately addressed in the design. Accordingly, the staff finds the system status consideration for the DPS is adequately addressed.

(8) Independence from the Protection Systems—IEEE Std 603, Sections 5.6 and 6.3

The staff evaluated whether the DPS functions are independent and diverse from the RPS and ESF actuation systems. The staff also evaluated whether ATWS mitigation systems are diverse from the RPS. The RPS uses the RTIF-NMS platform, while the ESF actuation systems uses the SSLC/ESF platform. The ATWS/SLC uses a safety ICP, while the DPS (nonsafety) uses a platform diverse from and independent of the safety platforms. This ensures that digital safety and nonsafety I&C systems are designed to minimize the potential for CCFs. The safety systems contain multiple redundant divisions to achieve high reliability. In accordance with IEEE Std 7-4.3.2, independence is provided between safety divisions and between safety and nonsafety systems to ensure that random single failures do not result in CCF that may affect multiple safety divisions.

As detailed in DCD Tier 2, Revision 9, Section 7.8.1.2.1, the DPS reactor trip functions provide a diverse means of reactor shutdown and serve as a backup to the RPS. This set of diverse protection logics for reactor scram, combined with the ATWS mitigation features, other diverse backup scram protection, and diverse ESF functions, provides the necessary diverse protections to meet the design requirements specified in the SRM to SECY-93-087 and BTP HICB-19. The applicant's D3 assessment in NEDO-33251, which is evaluated in Section 7.1.3 of this report, provides further information on diversity.

DCD Tier 2, Revision 9, Section 7.8.3, describes DPS conformance with IEEE Std 603. The DPS logic does not communicate with the RPS logic, and the DPS failure modes do not prevent the RPS from performing a reactor trip. The DPS cannot cause the RPS to initiate a reactor trip prematurely. Credible DPS failure modes cannot prevent the SSLC/ESF actuation system from initiating ECCS functions or performing barrier isolation functions. Additionally, credible failure modes cannot result in premature operation of these protection systems. The ATWS/SLC logic is designed to mitigate a failure of the normal reactor trip system to function and is diverse from and independent of the RPS. The ATWS/SLC logic platform is designed as a safety system with four independent divisions. The ATWS/ARI function is provided from the nonsafety DPS platform. Accordingly, the staff finds the independence from the protection systems consideration for the DPS is adequately addressed.

(9) Potential for Inadvertent Actuation

The staff evaluated whether the DPS design limits the potential for inadvertent actuation and challenges to safety systems. The DPS is designed as a highly reliable nonsafety system that meets the PRA requirements to minimize failures on demand and to minimize inadvertent operation. Consistent with the guidance in IEEE Std 603, the nonsafety DPS is designed to avoid adverse interaction with the protection systems with which the DPS interfaces. DCD Tier 2, Revision 9, Sections 7.8.1.2.1 and 7.8.1.2.2, describe features in the DPS that minimize inadvertent actuations. Initiation logic for the DPS (for reactor scram and ESF functions) is "energize to actuate." System-level operational and functional defenses, as described in DCD Tier 2, Revision 9, Section 7.8.2.2.1, include the following:

- Asynchronous operation of multiple protection divisions - timing signals are not exchanged among divisions
- Continuous cross-checking of redundant sensor inputs
- Automatic error checking on all multiplexed transmission paths
- Continuous self-test with alarm outputs in all system devices
- Automatic error detection (to permit early safe shutdown or bypass before common mode effects occur)
- Separation and independence requirements that protect against global effects resulting from such factors as EMI and thermal conditions

The staff finds that the DPS incorporates sufficient features to prevent inadvertent system actuation. The staff finds that the potential for inadvertent actuation consideration for the DPS is adequately addressed.

(10) Manual Initiation Capability

The staff evaluated whether the ATWS mitigation systems and DPS include the capability for initiation from the MCR. DCD Tier 2, Revision 9, Sections 7.8.1.1 and 7.8.1.2, provide details of the manual initiation capability for the ATWS/SLC, ATWS/ARI, and DPS. Manual capability is also included to mitigate failure of an ATWS logic processor. The MCR provides a manual bypass switch for this function. Switches in the MCR are also used to manually inhibit the ADS under ATWS conditions.

As a backup for the RPS, the DPS also provides the ability to initiate a manual scram from either hardwired switches or the DPS VDU. Additionally, manual initiation capability is provided in the DPS logic circuitry to initiate the diverse ECCS functions of the GDCS, SLC system, ICS, and ADS (SRVs and DPVs). The DPS also provides the ability to generate diverse manual ECCS actuation from the DPS VDU. Accordingly, the staff finds the manual initiation capability consideration for the DPS is adequately addressed.

(11) Completion of Protective Action

The staff evaluated whether the ATWS mitigation logic and DPS are designed such that, once initiated, the mitigation function will go to completion. DCD Tier 2, Revision 9, Section 7.8.1.1.1.1, describes the ATWS/SLC mitigation processor logic controls, which provide isolated hardwired contact closure outputs to the SLC system upon an initiation signal. Additionally, when an ATWS/ARI signal is received, the DPS generates an additional electrical signal to the RC&IS to initiate electrical insertion of all control rods. These actions are designed to ensure that mitigation functions for all AOOs (including ATWS) will go to completion. Accordingly, the staff finds the completion of protective action consideration for the DPS is adequately addressed.

(12) Diversity and Defense-in-Depth Analysis

The staff evaluated whether the DPS functions, credited with providing diversity are consistent with the assumptions of the applicant's D3 analysis. The staff reviewed NEDO-33251 and compared major aspects of the report with the DPS overview in DCD Tier 2, Revision 9, Section 7.8. NEDO-33251, Section 2.6, describes the I&C defense echelons and applications of the discrete I&C systems to each echelon. The DPS overview, defense echelons, CCF defenses, and system architecture match the descriptions of the DPS provided in DCD Tier 2, Revision 9, Sections 7.8.1 and 7.8.2. The staff finds that the DPS is consistent with the applicant's D3 LTR. Accordingly, the staff finds the D3 analysis consideration for the DPS is adequately addressed. Further information on diversity is provided in the applicant's D3 assessment (NEDO-33251), which is evaluated in Section 7.1.3 of this report.

7.8.3.2 *Evaluation of Diverse Instrumentation and Control System Compliance with Regulations and the SRM to SECY-93-087, Item II.Q*

GDC 1 requires quality standards and maintenance of appropriate records. The staff evaluated whether GDC 1 and 10 CFR 50.55a(a)(1) are adequately addressed for the safety ATWS mitigation logic and the nonsafety-related DPS , in accordance with SRP Appendix 7.1-A. SRP Appendix 7.1-A states that the staff review should confirm that the appropriate RGs and endorsed standards are identified as applicable for each I&C system important to safety. The staff's evaluation of conformance to RGs and standards for 10 CFR 50.55a(a)(1) and GDC 1 in Sections 7.1.1.3.3 and 7.1.1.3.6 of this report applies to the safety ATWS mitigation logic. In Section 7.1.1.3.3 of this report, the staff finds that the DCD has properly addressed compliance with the applicable RGs and standards for the safety systems including the ATWS mitigation systems. In DCD Tier 2, Revision 9, Sections 7.8.1 and 7.8.3.3, the applicant documented the conformance with the SRM to SECY 93-087, Item II.Q. The staff finds that 10 CFR 50.55a(a)(1) and GDC 1 for the ATWS mitigation logic are adequately addressed.

In DCD Tier 2, Revision 8, Section 7.8.3.4, the applicant updated the documentation with respect to the conformance with applicable RGs, as listed in SRP Appendix 7.1-A. The staff finds the clarification in DCD Section 7.8.3.4 acceptable. In Section 7.8.3.1, Item (2), of this report, the staff finds that the DCD has properly addressed the quality of the DPS.

10 CFR 50.55a(h) requires compliance with IEEE Std 603-1991 and the correction sheet dated January 30, 1995. The staff evaluated whether 10 CFR 50.55a(h) and IEEE Std 603 are adequately addressed for the safety DPS. The staff evaluation of IEEE Std 603 in Section 7.1.1.3.10 of this report applies to the safety DPS. DCD Tier 2, Revision 9, Section 7.8.1.1.1 states that a portion of the ATWS/SLC is implemented as safety. In Section 7.1.1.3.1

of this report, the staff finds that the DCD has properly addressed IEEE Std 603 compliance, including the DPS.

In DCD Tier 1, Revision 8, Section 2.2.15 (Table 2.2.15-1), the applicant added ATWS/SLC as an ICP in the IEEE Std 603 criterion applicability matrix. In DCD Tier 2, Revision 9, Section 7.8.3.1, the applicant documented the cross reference to the related DCD sections that address conformance to the IEEE Std 603 criterion. Based on this updated information, the staff finds that the requirements of 10 CFR 50.55a(h) and IEEE Std 603 are adequately addressed.

GDC 2 requires design bases for protection against natural phenomena. GDC 4 requires environmental and dynamic effect design bases. The staff evaluated whether GDC 2 and 4 are adequately addressed for the safety ATWS/SLC mitigation logic. SRP Appendix 7.1-A states that GDC 2 and 4 apply to all I&C safety systems. SRP Appendix 7.1-A for GDC 2 states that the design bases for protection against natural phenomena for I&C systems important to safety should be provided for the I&C system. SRP Appendix 7.1-A for GDC 4 states that the environmental and dynamic effects (e.g., missiles) design bases for I&C systems important to safety should be provided for each system described in Chapter 7 of the DCD. DCD Tier 2, Section 7.8.1.1.1, states that a portion of the ATWS/SLC is implemented as safety. However, neither DCD Tier 2, Table 7.1-1, nor DCD Tier 2, Section 7.8.3.2, specifies that GDC 2 and 4 apply to the safety-related ATWS/SLC mitigation logic. In RAI 7.1-99, Item G, the staff requested the applicant to address the applicability of the GDC to the ATWS/SLC. RAI 7.1-99 was being tracked as an open item in the SER with open items. In its response, the applicant revised DCD Tier 2, Revision 9, Table 7.1-1 and Section 7.8.3.2 to specify that GDC 2 and 4 apply to the safety-related ATWS/SLC mitigation logic. The staff finds that the response is acceptable since the applicant clarified that the ATWS/SLC Mitigation Logic conforms to GDC 2 and 4. Based on the applicant’s response, RAI 7.1-99 Item G is resolved.

DCD Tier 2, Revision 9, Table 3.2-1, identifies that the safety SLC safety electrical modules and cables, which includes the ATWS/SLC mitigation logic, are designed as seismic Category I systems. DCD Tier 2, Revision 9, Sections 3.10 and 3.11, describe the EQ programs for safety electrical and digital I&C equipment, which are evaluated in Chapter 3 of this report. DCD Tier 1, Revision 9, Table 3.8-1, Items 1 and 3, include ITAAC for the applicant to verify the EQ of safety electrical and digital I&C equipment. The evaluation of GDC 2 and GDC 4 in Section 7.1.1.3.6 of this report further addresses these topics and applies to the ATWS/SLC mitigation logic. Accordingly, based on the applicant’s identification of EQ programs consistent with the design bases for the ATWS/SLC mitigation logic and their verification in the ITAAC, the staff finds that the requirements of GDC 2 and 4 are adequately addressed for the safety-related ATWS/SLC mitigation logic.

GDC 24 requires that the protection system 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 of the protection system. GDC 24 requires that the interconnection of the protection and control systems be limited so as to assure that safety is not significantly impaired. The staff evaluated whether GDC 24 is adequately addressed for the DPS. Appendix 7.1-A to the SRP states that GDC 24 is addressed for safety systems by conformance to IEEE Std 603, Sections 5.1, 5.6, 5.12, 6.3, 6.6, and 8, particularly Sections 5.6 and 6.3. DCD Tier 2, Revision 9, Table 7.1-1, identifies that GDC 24 applies to the DPS. DCD Tier 2, Revision 9, Section 7.8, describes the conformance of the DPS to IEEE Std 603, Sections 5.6 and 6.3, which are evaluated in Section 7.1.1.3.10 in this report. For the nonsafety DPS, DCD Tier 2, Revision 9, Sections 7.1.6.6.1.7 and 7.1.6.6.1.19, describe conformance with IEEE Std 603, Sections 5.6 and 6.3. These sections state that the Q-DCIS protection systems are separate and independent of the nonsafety control systems, in accordance with GDC 24, and that any failure of nonsafety systems does not affect safety protection systems or prevent them from performing their safety functions. Section 7.1.1.3.10 of this report evaluates Sections 5.6 and 6.3 of IEEE Std 603. DCD Tier 1, Revision 9, Table 2.2.15-2, includes the DAC/ITAAC for verifying that the applicable I&C systems' design is completed in compliance with IEEE Std 603, including Sections 5.6 and 6.3. Accordingly, the staff finds that the requirements of GDC 24 are adequately addressed for the DPS.

The staff evaluated whether GDC 13 and 19 are adequately addressed for the DPS. GDC 13 requires providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions to assure adequate safety. GDC 13 also requires providing appropriate controls to maintain these variables and systems within prescribed operating ranges. GDC 19 requires control room functionality, control room habitability, and remote shutdown capability. Section 7.1.1.3.6 of this report evaluates GDC 19 with the exception of support functions necessary for operating the reactor. SRP Section 7.8 identifies that GDC 13 and 19 are addressed by the review of the DPS status information, manual initiation capabilities, and control capabilities. DCD Tier 2, Revision 9, Section 7.8.5, specifies the status information of the safety ATWS mitigation logic. DCD Tier 2, Revision 9, Section 7.8.1.1, specifies the automatic and manual initiation and control capabilities of the safety ATWS/SLC mitigation logic. DCD Tier 2, Revision 9, Section 7.8.1.2, specifies the automatic initiation, controls, status information, and manual initiation and control capabilities of the DPS. In combination with the following identified interrelated processes to complete the design of the monitoring capability and control room controls for the DPS, the staff finds these monitoring capabilities and controls acceptable.

NEDE-33226P and NEDE-33245P, as part of a software life cycle process, define a process by which plant performance requirements under various operational conditions will be specified, implemented, and tested. DCD Tier 1, Revision 9, Section 3.2, includes the DAC/ITAAC for verifying that the software plans are developed and implemented consistent with this process and produce acceptable design outputs. DCD Tier 1, Revision 9, Section 3.3, includes the DAC/ITAAC for implementing an HFE design process, which includes the design and verification of controls and information displays for monitoring variables and systems in the control room. These verifications apply to the DPS and include verifications of the controls for manual initiation and control of the DPS functions necessary to support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Accordingly, based on the identified monitoring capabilities and control room controls, the defined processes for completing their design, and their verification in the Revision 9 DAC/ITAAC, the staff finds that the requirements of GDC 13 and 19 are adequately addressed for the DPS.

The staff evaluated whether 10 CFR 50.62 is adequately addressed for the DPS. As required by 10 CFR 50.62, BWR plants must have (1) an ARI system that is diverse from the RPS and the ARI must be designed to perform its function in a reliable manner and be independent from the RPS [10 CFR 50.62(c)(3)], (2) an SLC system whose initiation must be automatic and which must be designed to perform its function in a reliable manner [10 CFR 50.62(c)(4)], and (3) an automatic recirculation pump trip (RPT) [10 CFR 50.62(c)(5)]. Because the ESBWR design does not have a recirculation pump, 10 CFR 50.62(c)(5) does not apply. SRP Table 7.1 identifies that 10 CFR 50.62 applies to the DPS (DCD Section 7.8). DCD Tier 2, Revision 9,

Table 7.1-1, identifies that 10 CFR 50.62 applies to the relevant safety and nonsafety I&C systems.

For 10 CFR 50.62(c)(3), DCD Tier 2, Revision 9, Section 7.8.1.1, describes the diverse ATWS mitigation logic, which includes an ARI, SCRRI/SRI, and diverse scram. DCD Tier 2, Revision 9, Section 7.8.1, states that the ARI uses the three sets of air header dump valves of the CRD system to hydraulically scram the plant. DCD Tier 2, Revision 9, Figure 4.6-8, shows that these valves are redundant to the scram air header dump valves. DCD Tier 2, Section 7.8.1.1.2, states that the ARI logic resides in the DPS, which is separate and independent from the Q-DCIS with diverse hardware and software. The RPV pressure and level input sensors for the ARI logic are independent and separate from the sensors used in the Q-DCIS. Based on the above, the staff finds that the diverse ATWS mitigation logic includes the ARI functions required by 10 CFR 50.62(c)(3).

For 10 CFR 50.62(c)(4), DCD Tier 2, Revision 9, Section 7.8.1.1.1, describes the safety ATWS mitigation logic, which includes the automatic initiation of SLC boron injection and feedwater runback. DCD Tier 2, Section 7.8.1.1.1, also states that the safety ATWS mitigation logic processors are separate and diverse from RPS circuitry and use discrete programmable logic devices for ATWS mitigation logic processing. DCD Tier 2, Revision 9, Section 7.8.3, states that the safety ATWS mitigation logic is diverse from and independent of the RPS. Based on the above, the staff finds that the safety ATWS mitigation logic includes the SLC functions required by 10 CFR 50.62(c)(4).

As noted above, 10 CFR 50.62(c)(5) requires an automatic RPT because the BWR uses forced core flow circulation. Because the ESBWR design uses natural circulation, there are no recirculation pumps to be tripped. The automatic FWRB feature is implemented to provide a reduction in water level, core flow, and reactor power, similar to the RPT in a forced circulation plant.

The staff evaluated the design features that provide diversity of each ATWS mitigation logic from the RPS and its functioning in a reliable manner. The design uses the FMCRD design with both hydraulic and electrical means to achieve shutdown. The use of this design eliminates the CCF potentials of the locking-piston CRD by eliminating the scram discharge volume and by having an electrical motor run-in diverse from the hydraulic scram feature. This feature allows rod run-in, if scram air header pressure is not exhausted because of a postulated common cause electrical failure and simultaneous failure of the ARI system, thus satisfying the intent of 10 CFR 50.62.

The ATWS mitigation functions use diverse control logics from the primary protection system. The safety portions of the ATWS mitigation logic, which provides an alternate means of emergency plant shutdown via soluble boron injection by the SLC system, use independent logic controllers instead of a software-based platform.

DCD Tier 2, Revision 9, Section 7.8.1, discusses CCF defenses. The DPS triple redundant design employs a hardware and software platform which differs from the primary protection system platforms (i.e., RTIF-NMS, ECCS/ESF, and ICP). The DPS is diverse from and independent of the primary protection systems. As part of the D3 evaluation, the applicant performed a review to assess the impact of a digital protection system CCF on events discussed in DCD Tier 2, Revision 9, Chapter 15. NEDO-33251 documents the coverage of the DPS with respect to the DCD Tier 2, Chapter 15, events and discusses the backup functions provided by the DPS for mitigation of DCD Tier 2, Chapter 15, events. Accordingly, the staff finds that the ATWS mitigation design includes an appropriate set of functions. As described in Section 7.8.3.1, Item (8) above, in this report, the staff finds that the separation and independence design features of the RTS are not compromised by the ATWS mitigation system design. Where isolation devices are provided in the RTS to support ATWS mitigation interfaces, the isolation devices are applied and qualified to the guidelines of BTP HICB 7-11. Based on the above, the staff finds that ATWS mitigation logic is acceptably diverse from the RPS and provides reasonable assurance of functioning in a reliable manner.

Based on the diverse ATWS mitigation logic, including the ARI functions required by 10 CFR 50.62(c)(3); the safety ATWS mitigation logic, including the SLC functions required by 10 CFR 50.62(c)(4); and both ATWS mitigation logics having acceptable diversity from the RPS and providing reasonable assurance of functioning in a reliable manner, the staff finds that 10 CFR 50.62 is adequately addressed for the DPS.

The staff evaluated whether the guidelines of the SRM to SECY-93-087, Item II.Q, are adequately addressed for the DPS. Section 7.1.1.3.7 of this report evaluates the SRM to SECY-93-087, Item II.Q, which applies to the DPS. DCD Tier 2, Revision 9, Table 7.1-1, identifies that the SRM to SECY-93-087, Item II.Q, applies to the DPS. NEDO-33251 provides the primary assessment of conformance to the guidelines of SRM to SECY-93-087, Item II.Q, along with BTP HICB-19. Section 7.1.3 of this report documents the staff's evaluation of the D3 assessment, which finds the following:

- The applicant has analyzed each postulated CCF for each event that is evaluated in the accident analysis section of DCD Tier 2, Revision 9, Chapter 15, and the applicant has demonstrated adequate diversity within the DCIS design for each of these events.

- The proposed DPS has sufficient quality to perform the necessary function under the associated event conditions.

- A set of displays and controls located in the MCR can provide for manual system-level actuation of critical safety functions. The displays and controls are independent and diverse from the Q-DCIS.

Sections 7.1.1.3.7 and 7.1.3 of this report document the staff's evaluation of the I&C system in response to the SRM to SECY-93-087.

The staff evaluated whether the requirements of 10 CFR 52.47(b)(1) are adequately addressed. DCD Tier 1, Revision 9, Section 2.2.14, contains the 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 references the design certification has been constructed and will operate in accordance with the design certification, the Atomic Energy Act, and the Commission's rules and regulations.

7.8.4 Conclusion

Based on the review of information in DCD Tier 1, Revision 9, Sections 2.2.14 and 2.2.15; and DCD Tier 2, Revision 9, Section 7.8; and NEDO-33251, the staff concludes that the DPS design meets the applicable regulatory requirements of 10 CFR 50.55a(a)(1); 10 CFR 50.55a(h); GDC 1, 13, 19, and 24; 10 CFR 52.47(b)(1); 10 CFR 50.62; and the guidelines of the SRM to SECY-93-087.

8.0 ELECTRIC POWER

8.1 Introduction

The economic simplified boiling-water reactor (ESBWR) design, as presented, does not require Class 1E alternating current (ac) electrical power, except that provided by the Class 1E direct current (dc) batteries and their inverters, to accomplish the plant's safety-related functions.

Two independent offsite power sources provide reliable power for the plant's auxiliary loads, such that any single active failure can affect only one power source and cannot propagate to the alternate power source.

The onsite ac power system consists of Class 1E and non-Class 1E power systems. The two offsite power systems provide the normal preferred and alternate preferred ac power to the onsite power systems. In the event of a total loss of offsite power sources, two onsite independent nonsafety-related standby diesel generators (DGs) provide power to the plant investment protection (PIP) nonsafety-related loads and safety-related loads through safety-related isolation power centers (IPCs). Four independent safety-related 480-volt (V) ac divisions (IPC buses) provide power for the safety-related 250-V dc systems and uninterruptible power supply (UPS) systems.

Table 8.1 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)", March 2007 (SRP), lists the applicable regulatory requirements, guidance, and associated acceptance criteria that apply to electric power systems. Sections 8.2, 8.3.1, 8.3.2, and 8.4 of this report analyze the application's conformance with the regulatory requirements and associated acceptance criteria listed in SRP Table 8.1 regarding electric power systems.

8.2 Offsite Power System

8.2.1 Regulatory Criteria

The offsite power system includes two physically independent circuits capable of operating independently of the onsite standby power sources. The review by the staff covers the information, analyses, and documents for the offsite power system and the stability studies for the electrical transmission grid. In general, the preferred power system is acceptable when it provides two separate circuits from the transmission network to the onsite Class 1E power distribution system, when adequate physical and electrical separation exist, and when the system has the capacity and capability to supply power to all safety loads and other required equipment.

The acceptance criteria for assessing the sufficiency of the offsite power system design are based on meeting the following relevant requirements:

- General Design Criterion (GDC) 5, "Sharing of structures, systems, and components," 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," as it relates to sharing structures, systems, and components (SSCs) of the preferred power systems;

- GDC 17, “Electric power systems,” as it relates to the preferred power system’s (1) capacity and capability to permit functioning of SSCs important to safety, (2) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies, (3) physical independence, and (4) availability;

- GDC 18, “Inspection and testing of electric power systems,” as it relates to the inspection and testing of the offsite power systems;

- GDC 33, “Reactor coolant makeup,” 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,” as they relate to the operation of the offsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in GDC 33, 34, 35, 38, 41, and 44 are accomplished;

- 10 CFR 50.63, as it relates to an alternate ac power source (as defined in 10 CFR 50.2) provided for safe shutdown in the event of a station blackout (SBO) (not a design-basis accident [DBA]);

- 10 CFR 50.65(a) (4) (the maintenance rule), as it relates to the assessment and management of the increase in risk that may result from proposed maintenance activities before performing such activities.

Note that subsequent to the issuance of SRP Section 8.2, Revision 4, the staff determined that GDC 2, “Design bases for protection against natural phenomena,” and GDC 4, ”Environmental and dynamic effects design bases,” are not applicable to the offsite power system. The staff determination is documented in a January 23, 2009 e-mail from Thomas Bergman to Russ Bell, “NRC response to GDC 2, 4 and 5 one-pager.”

SRP Sections 8.1 and 8.2, and Appendix A to Section 8.2, explain how an application can meet the above regulations, in part, if the application uses the NRC-endorsed methodologies and technical positions found in the following:

- Regulatory Guide (RG) 1.32, “Criteria for Power Systems for Nuclear Power Plants,” Revision 3, issued March 2004.

- RG 1.155, “Station Blackout,” issued August 1988.

- RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2, issued March 1997.

- RG 1.182, “Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants,” Revision 0, issued May 2000.

- RG 1.204, “Guidelines for Lightning Protection of Nuclear Power Plants,” issued November 2005.

- Branch Technical Position (BTP) 8-3, “Stability of Offsite Power Systems.”

- BTP 8-6, “Adequacy of Station Electric Distribution System Voltages.”

- BTP Power Systems Branch (PSB) -1, “Adequacy of Station Electric Distribution System Voltages,” Revision 3, issued April 1996.

- BTP Instrumentation and Control Systems Branch (ICSB) -11, “Stability of Offsite Power Systems,” Revision 3, issued April 1996.

- SECY-90-016, “Evolutionary Light Water Reactor Certification Issues and Their Relationships to Current Regulatory Requirements,” issued 1990.

- SECY-91-078, “EPRI’s Requirements Document and Additional Evolutionary LWR Certification Issues,” issued 1991.

- SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” issued 1994.

- SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs,” issued 1995.

8.2.2 Summary of Technical Information

Two independent offsite power supply systems supply power to the site, which include the normal preferred power supply (PPS) through the unit auxiliary transformers (UATs) and the alternate PPS through the reserve auxiliary transformers (RATs). The offsite system is designed and located to minimize the likelihood of simultaneous failure during a design basis accident (DBA) and under adverse environmental conditions.

The main generator normally provides power to the onsite power system through the two UATs as the normal preferred source. When the main generator is not available, the generator output breaker is opened, and the onsite auxiliary power is supplied from the switchyard by backfeeding through UATs as the normal preferred source. When the normal preferred source is not available, the plant auxiliary power is supplied from the switchyard through the two RATs as the alternate preferred source. In addition, two nonsafety-related onsite standby DGs supply power to selected loads in the event of the loss of two offsite preferred sources.

The main generator is connected to the offsite power system by three single-phase, step-up transformers. If the main generator is lost, an auto-trip of the generator breaker maintains the power to the onsite system without interruption from the PPS. A spare single-phase, step-up transformer is available, in case one of the main step-up transformers fails. The main step-up transformers are within the onsite power system.

Unit synchronization normally occurs through the onsite main generator circuit breaker, with the offsite switchyard circuit breaker supplying the normal preferred power source. Both the main generator circuit breaker and the normal PPS circuit breakers are equipped with dual trip coils and a redundant protective relaying logic scheme, and redundant nonsafety-related 125-V dc power systems supply control power to the circuit breaker.

The UATs consist of two three-phase transformers. Each UAT provides normal preferred power to two power generation (PG) buses and one PIP bus. The RATs consist of two three-phase transformers fed from the second offsite power source. Each RAT provides alternate preferred power to the plant’s two PG buses and one PIP bus in the event of a UAT failure. The main step-up transformers, UATs, and RATs are designed and manufactured to withstand the mechanical and thermal stresses produced by external short circuits, and meet the corresponding requirements of Institute for Electrical and Electronics Engineers (IEEE) Standard (Std) C57.12.00, "Standard General Requirements for Liquid-Immersed Distribution, Power, and Regulating Transformers." The main power transformers, UATs, and RATs have protective devices for overcurrent, differential current, ground overcurrent, and sudden overpressure.

An onsite generator circuit breaker can interrupt the maximum fault current. The generator circuit breaker allows the generator to be taken off the power supply system, and the switchyard offsite power system backfeeds power to the onsite ac power systems. Startup power is normally provided through UATs from the switchyard.

Protective relaying schemes used to protect the offsite power supply system in the switchyard are redundant and equipped with backup protection features, and the circuit breakers are equipped with dual trip coils. Each redundant protection circuit is powered from its redundant dc power supply and connected to a separate trip coil. Equipment and cabling associated with each redundant system are physically separate.

The offsite power system of the ESBWR plant is based on the following design bases:

- In the event of failure of the normal PPS system, the alternate PPS system remains available.

- The normal PPS system and the alternate power supply system are electrically independent and are physically separated from each other. Separate transmission lines, each capable of supplying the shutdown loads, feed the normal and the alternate PPS systems.

- The switching station to which the main offsite circuit is connected has two buses, arranged such that any incoming transmission line can be isolated by tripping a circuit breaker, without affecting another line, and faults of one bus are isolated without interrupting service to any line.

- Circuit breakers are sized and designed in accordance with IEEE Std C37.06, "AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis—Preferred Ratings and Related Required Capabilities."

- Concrete barriers with a fire rating of 3 hours are used among the main transformers, the UATs, and the RATs, including the containment and collection of transformer oil.

- Cables associated with the normal and alternate PPS systems are routed separately and in separate raceways, apart from each other and onsite power system cables. However, they may share a common underground duct bank, as indicated below.

- Associated control, instrumentation, and miscellaneous power cables of the alternate preferred circuit are routed in separate raceways, if they are located underground in the same duct bank as cables associated with the normal PPS system between the switchyard and the onsite power systems.

- Interface protocols shall be established between the control room and the transmission operator, in accordance with the interconnection service agreement.

- Cables associated with the alternate preferred supply systems are routed in trenches within the switchyard, separate from cables associated with the normal preferred supply systems.

- A transmission system reliability and stability review of the site-specific configuration to which the plant is connected will determine the reliability of the offsite power supply system and verify that it is consistent with the probability risk analysis of Chapter 19 of the ESBWR design control document (DCD).

- The design provides for an auto-disconnect of the high side of a failed UAT through protective relaying to UAT input circuit breakers and RAT motor-operated disconnects (MODs).

- A station grounding system, consisting of a ground mat below grade at the switchyard that is connected to the foundation's embedded loop grounding system, serves the entire power block and associated buildings.

8.2.3 Staff Evaluation

The following paragraphs analyze compliance with the GDC and consistency with the RGs and other SRP guidance:

- GDC 5

  Because the ESBWR plant is designed as a single-unit plant, GDC 5 and RG 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants," Revision 1, issued January 1975, are not applicable.

- GDC 17

  With regard to GDC 17, the ESBWR plant design does not require offsite or diesel-generated ac power for 72 hours after an abnormal event. Safety-related dc power supports passive core cooling and containment safety-related functions. In Request for Additional Information (RAI) 14.3-394 S01, the staff asked the applicant to provide an interface requirement for demonstrating the capacity and capability of the offsite power system. In its response, the applicant incorporated the PPS definition into DCD Tier 2, Revision 6, Chapter 8, per IEEE Std 765, "IEEE Standard for Preferred Power Supply (PPS) for Nuclear Power Generating Stations," issued June 2002. The PPS consists of the normal preferred and alternate preferred power sources and includes those portions of the offsite power system and the onsite power system required for power flow from the offsite transmission system to the safety-related IPC incoming line breakers. Additionally, the applicant revised DCD Tier 2, Revision 5, Chapter 8, to clarify that GDC 17 applies to the entire PPS. The applicant added inspection, test, analysis, and acceptance criteria (ITAAC) to DCD Tier 1, Revision 6, Section 2.13.1, to address capacity, capability, and the physical and electrical separation of the normal preferred and alternate PPSs. The staff agrees with the applicant regarding the boundary of the PPS; i.e., at the incoming line breaker of the IPC bus. The staff confirmed that Revision 6 of the DCD includes the changes described above. Based on the applicant's response, RAI 14.3-394 S01 regarding GDC 17 is resolved. Chapter 14 of this report discusses RAI 14.3-394 S01 regarding ITAAC.

  The staff reviewed the offsite power system design-basis requirements in DCD Tier 2, are resolved Section 8.2.3, and finds that these design-basis requirements include (1) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies, (2) physical independence, and (3) for availability. These design bases are consistent with the requirements of GDC 17 and therefore the staff finds offsite power system design-basis requirements acceptable. Based on the above, the staff finds that the ESBWR standard design complies with GDC 17 with respect to two independent and separate offsite power sources.

For the degraded and overvoltage protection evaluation, refer to Section 8.3.1 of this report.

- GDC 18

  In RAI 8.2-15, the staff asked the applicant to discuss how it will comply with GDC 18 for the portion of the offsite system (MOD, UAT high-side breaker, UAT, PIP bus, 6.9/0.48-kilovolt (kV) transformer for normal PPS, and MOD, RAT, PIP bus, 6.9/0.48-kV transformer for the alternate PPS) that is within the DCD's scope. In response, the applicant stated that the above-mentioned equipment is part of the PPS and will be covered by procedures for testing and maintenance, as described by combined license (COL) Information Items 13.5-2-A and 13.5-6-A. These procedures will state the frequency requirements for tests and maintenance. Based on the applicant's response, RAI 8.2-15 is resolved. The staff finds that the ESBWR standard plant design complies with the requirements of GDC 18.

- GDC 33, 34, 35, 38, 41, and 44

  The potential risk contribution of a design-basis event is minimized because a passive reactor design does not require ac power sources for such events. Passive reactor designs incorporate passive safety-related systems for core cooling and containment integrity and, therefore, do not depend on the electric power grid connection and grid stability for safe operation. They are designed to automatically establish and maintain safe shutdown conditions after design-basis events for 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. Therefore, the ESBWR offsite power system design is not required to meet the requirements of GDC 33, 34, 35, 38, 41, and 44.

- 10 CFR 50.63

  Section 15.5.5 of this report provides the staff's evaluation of the ESBWR coping capability with an ac-independent approach and finds it to be acceptable. Therefore, the ESBWR design also meets the requirements of 10 CFR 50.63. The EBSWR SBO evaluation is further discussed in Section 8.4.2.1 of this report.

- 10 CFR 50.65(a) (4)

  The maintenance rule requires a licensee to evaluate grid reliability as part of the maintenance risk assessment before performing grid-risk-sensitive maintenance activities. As described in DCD Tier 2, Revision 9, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

- RG 1.32

  The offsite power system is not Class 1E. Therefore, RG 1.32 is not applicable to the ESBWR offsite power system.

- RG 1.155

  No ac power is required to achieve safe shutdown for the ESBWR. The ESBWR uses battery power to achieve and maintain safe shutdown. The safety-related batteries have sufficient stored capacity, without their chargers, to independently supply the safety-related loads continuously for 72 hours. This meets the ac independent coping capability guidelines of RG 1.155. (The SBO evaluation is further discussed in Sections 8.4.2.1 and 15.5.5 of this report.)

- RG 1.160

  As described in DCD Tier 2, Revision 9, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

- RG 1.182

  As described in DCD Tier 2, Revison 8, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

- RG 1.204

  Section 8.3.1 of this report discusses this topic. The design is consistent with the guidance of RG 1.204.

- BTP 8-3 and BTP ICSB-11

  This topic is site-specific and will be addressed by the COL applicant (COL Information Item 8.2.4-9-A). See Section 8.2.4 of this report.

- BTP 8-6 and BTP PSB-1

  This topic is addressed in DCD Tier 1, Revision 9, Section 2.13.1, and is evaluated in Section 14.3.6 of this report.

- SECY-90-016

  This paper contains the Commission's approval for the evolutionary advanced light-water reactors (ALWRs) to have an alternate ac power source of diverse design capable of powering at least one complete set of normal shutdown loads to cope with SBO. This topic is not applicable to the ESBWR design, since no ac power is required to achieve safe shutdown, as discussed in the SBO evaluation in Section 8.4.2.1 of this report.

- SECY-91-078

  This paper relates to (1) the inclusion of an alternate power source for nonsafety-related loads in an evolutionary plant design and (2) at least one offsite circuit to each redundant safety division, to be supplied directly from one of the offsite power sources with no intervening nonsafety buses, in such a manner that the offsite source can power the safety buses if any nonsafety bus fails. As discussed in Section 8.4.2.2 of this report, the ESBWR design meets recommendation 1, above. The ESBWR design does not have to meet recommendation 2, because the design does not rely on active systems for safe shutdown.

- SECY-94-084 and SECY-95-132

  These papers relate to the use of alternate ac power sources and the application of RTNSS at ALWRs provided with passive safety systems. An alternate ac power source is not required to achieve safe shutdown in a passive design as discussed in Section 8.4.2.1 of this report as part of the SBO evaluation. The portions of the offsite power system which have RTNSS functions; i.e. the 6.9 kV PIP buses, are discussed in Sections 8.4.2.3 and 22 of this report.

8.2.4 Combined License Unit-Specific Information

The applicant stated that a COL applicant will address the following items:

- 8.2.4-1-A Transmission System Description
- 8.2.4-2-A Switchyard Description
- 8.2.4-3-A Normal Preferred Power
- 8.2.4-4-A Alternate Preferred Power
- 8.2.4-5-A Protective Relaying
- 8.2.4-6-A Switchyard DC Power
- 8.2.4-7-A Switchyard AC Power
- 8.2.4-8-A Switchyard Transformer Protection
- 8.2.4-9-A Stability and Reliability of the Offsite Transmission Power Systems
- 8.2.4-10-A Interface Requirements

These COL items identify ten items related to the offsite power system that the COL applicant will address. The staff agrees that these items are site specific and therefore appropriately addressed by the COL applicant. In addition, these COL items for the offsite power system, provided they are adequately addressed, provide reasonable assurance that the offsite power system meets the requirements of GDC 17. Therefore, the staff finds these COL information items acceptable.

8.2.5 Conclusions

The staff considers the applicant’s description to be acceptable on the basis that it provides sufficient information on the scope of the offsite system and that the system meets the requirements discussed above. Further, the staff finds the design-bases requirements for the offsite power system to be acceptable. Therefore, the staff concludes that the design of the offsite power system for the ESBWR is acceptable and meets the requirements of GDC 17 and 18.

8.3 <u>Onsite Power System</u>

8.3.1 Alternating Current Power System

8.3.1.1 *Regulatory Criteria*

The onsite ac power system consists of a Class 1E and a non-Class 1E system that, in conjunction, provide reliable ac power to the electrical loads of the various Class 1E and non-Class 1E systems. These loads enhance an orderly shutdown under emergency (not accident) conditions. Additional loads for investment protection can be manually loaded on the standby power supply systems. The staff’s review covers the descriptive information, analyses, and referenced documents for the ac onsite power system, as well as the applicable recommendations from the SRP and appropriate standards and design criteria, as follows:

- GDC 2, as it relates to SSCs of the ac power system being capable of withstanding the effects of natural phenomena without losing the ability to perform their safety functions

- GDC 4, as it relates to SSCs of the ac power system being capable of withstanding the effects of missiles and environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC 5, as it relates to the sharing of SSCs of the ac power systems

- GDC 17, as it relates to the onsite ac power system's (1) capacity and capability to permit functioning of SSCs important to safety, (2) independence, redundancy, and testability to perform its safety function, assuming a single failure, and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network

- GDC 18, as it relates to the inspection and testing of the onsite power systems

- GDC 33, 34, 35, 38, 41, and 44, as they relate to the operation of the onsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in these GDC are accomplished

- GDC 50, "Containment Design Basis," as it relates to the design of containment electrical penetrations containing circuits of the ac power system and the capability of electric penetration assemblies in containment structures to withstand a loss-of-coolant accident (LOCA) without loss of mechanical integrity, as well as the external circuit protection for such penetrations

- 10 CFR 50.63, as it relates to the establishment of a reliability program for emergency onsite ac power sources and the use of the redundancy and reliability of DG units as a factor in limiting the potential for SBO events

- 10 CFR 50.65(a) (4), as it relates to the assessment and management of the increase in risk that may result from proposed maintenance activities before performing such activities

SRP Section 8.3.1 provides guidance on how an application can meet the above regulations.

- GDC 5 is satisfied as it relates to the sharing of SSCs of the ac power system and the following guidelines:
  - RG 1.32, as it relates to the sharing of SSCs of the Class 1E power system at multiunit stations
  - RG 1.81, as it relates to the sharing of SSCs of the ac power system, Regulatory Positions C.2 and C.3

- GDC 17 is satisfied, as it relates to the onsite ac power system's (1) capacity and capability to permit functioning of SSCs important to safety, (2) independence, redundancy, and testability to perform its safety function, assuming a single failure, and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network, in conformance with the following guidelines:

  - RG 1.6, “Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6),” issued March 1971, as it relates to the independence of the onsite ac power system, Regulatory Positions D.1, D.2, D.4, and D.5
  - RG 1.9, “Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants,” Revision 3, issued July 1993 (also IEEE Std 387, “IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations,” issued 1984)
  - RG 1.32, as it relates to design criteria for onsite ac power systems (also IEEE Std 308, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations”)
  - RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems,” Revision 2, issued November 2003, as it relates to the application of the single-failure criterion to safety systems (also IEEE Std 279, “IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations,” and IEEE Std 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations”)
  - RG 1.75, “Criteria for Independence of Electrical Safety Systems,” Revision 3, issued February 2005, as it relates to the onsite ac power system (also IEEE Std 384, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits”)
  - RG 1.153, “Criteria for Safety Systems,” Revision 1, issued June 1996, as it relates to criteria for electrical portions of safety-related systems (also IEEE Std 603)
  - RG 1.155, as it relates to the use of onsite emergency ac power sources for SBO
  - RG 1.204, as it relates to the lightning and surge protection for the onsite ac power system (also IEEE Std 665, “IEEE Guide for Generating Station Grounding”; IEEE Std 666, “IEEE Design Guide for Electric Power Service Systems for Generating Stations”; IEEE Std 1050, “IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations”; and IEEE Std C62.23, “IEEE Application Guide for Surge Protection of Electric Generating Plants”)

- GDC 18 is satisfied, as it relates to the testability of the onsite ac power system and the following guidelines:

  - RG 1.32, as it relates to the capability for testing the onsite ac power system (also IEEE Std 308)
  - RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems,” issued May 1973, with respect to indicating the bypass or inoperable status of portions of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the system it actuates to perform their safety-related functions
  - RG 1.118, “Periodic Testing of Electric Power and Protection Systems,” Revision 3, issued April 1995, as it relates to the capability for testing the onsite ac power system (also IEEE Std 338, “IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems”)

- RG 1.153, as it relates to the onsite ac power system (also IEEE Std 603)

- GDC 33, 34, 35, 38, 41, and 44 are satisfied as they relate to the operation of the onsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in these GDC are accomplished.

- GDC 50, as it relates to the design of containment electrical penetrations containing circuits of the ac power system, is satisfied, in conformance with the guidelines of RG 1.63, “Electric Penetration Assemblies in Containment Structures for Nuclear Plants,” issued September 1987, as it relates to the capability of electric penetration assemblies in containment structures to withstand a LOCA without loss of mechanical integrity and the external circuit protection for such penetrations, as well as to ensuring that electrical penetrations will withstand the full range of fault current (minimum to maximum) available at the penetration (see also IEEE Std 242, “IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems”; IEEE Std 317, “IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations”; and IEEE Std 741, “IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations”)

- 10 CFR 50.63, as it relates to using the redundancy and reliability of DG units as a factor in limiting the potential for SBO events, with acceptance based on meeting the following specific guidelines:
  - RG 1.9, as it relates to the adequacy of the DG surveillance criteria provided to attain and maintain the target reliability levels of DG units
  - RG 1.155, as it relates to using the reliability of emergency onsite ac power sources as a factor in determining the coping duration for SBO (noting that SRP Section 8.4 reviews this determination in detail) and the establishment of a reliability program for attaining and maintaining source target reliability levels.

- 10 CFR 50.65(a)(4), as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing such activities, with acceptance based on meeting the following specific guidelines:
  - RG 1.160, as it relates to the effectiveness of maintenance activities for onsite emergency ac power sources, including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase loss-of-offsite-power (LOOP) frequency, or reduce the capability to cope with a LOOP or SBO)
  - RG 1.182, as it relates to implementing the provisions of 10 CFR 50.65(a)(4) by endorsing Section 11 of Nuclear Management and Resources Council 93-01, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” dated February 22, 2000.

8.3.1.2 *Summary of Technical Information*

The main onsite ac power system is a non-Class 1E system. During PG mode, the turbine generator normally supplies electric power to the plant’s onsite auxiliary loads through UATs. The plant will be designed to sustain a load rejection from 100-percent power, with the turbine generator supplying the plant’s house loads.

During plant startup, shutdown, and maintenance, the generator breaker will be opened. Under this condition, the PPS systems supply the main ac power from the transmission switchyard through the two UATs. Each UAT supplies power to one of the two separate load groups of the onsite ac power system, each with a redundant RAT for backup as the alternate PPS system.

The onsite ac power system consists of the PPS and the onsite standby ac power supply system, as well as the electrical distribution systems. The preferred power systems consist of buses of two different voltage levels (i.e., 13.8-kV PG nonsafety-related buses and 6.9-kV PIP nonsafety-related buses).

- PG nonsafety-related buses have connections to the normal or alternate offsite PPS systems through the UATs or RATs, respectively. The PG nonsafety-related buses are the 13.8-kV unit auxiliary switchgear and associated lower voltage load buses.

- PIP nonsafety-related buses have connections to the normal or alternate offsite PPS systems through the UATs or RATs, respectively, with backup from the standby onsite ac power supply system. Reverse power relaying prevents backfeed to the standby onsite ac power supply system. The PIP nonsafety-related buses are the 6.9-kV PIP buses and associated lower voltage load buses, exclusive of the safety-related IPC 480-V ac buses.

The PG nonsafety-related buses feed nonsafety-related loads that are required exclusively for unit operation and are usually powered from the normal preferred power through the UATs. If the normal PPS system is not available, these buses are automatically transferred to the alternate PPS system through a fast transfer scheme. On restoration of the normal PPS, these buses are manually or automatically transferred to the normal PPS system, depending on the selection of manual or automatic transfer mode, or they remain powered from the alternate preferred power source.

The PIP nonsafety-related buses feed nonsafety-related loads generally required to remain operational at all times or when the unit is shut down. In addition, the PIP buses supply ac power to the safety-related buses, which are IPC buses. The PIP buses are backed up by the onsite standby power supply systems. These buses can also be powered from the alternate PPS through an automatic bus transfer, in the event that the normal PPS is unavailable. On restoration of the normal PPS, these buses are manually or automatically transferred to the normal PPS system, depending on the selection of manual or automatic transfer mode, or they remain powered from the alternate preferred power source.

8.3.1.2.1 Electrical Distribution System

Medium-Voltage Alternating Current Power Distribution System

The medium-voltage ac power distribution system consists of four 13.8-kV PG buses and two 6.9-kV PIP buses. The UATs and RATs, at 13.8 kV and 6.9 kV, supply the medium-voltage ac power to the PG and PIP buses. Each of the four PG buses is powered from one of the two UATs, or if the UATs are unavailable, from one of the two RATs. The incoming circuit breakers for each PG bus are electrically interlocked to prevent simultaneous connections of UATs and RATs to the PG bus.

Two 6.9-kV PIP buses (PIP-A and PIP-B) provide power for the nonsafety-related PIP loads, so that the unit can remain operational at all times or during shutdown. In addition, the PIP buses supply ac power to the safety-related load through the IPCs. PIP-A and PIP-B buses are each backed up by a separate onsite standby ac power supply system. Each PIP bus will be powered from the normal PPS system through the UAT. If the normal PPS system is unavailable, these buses will be automatically transferred to the alternate PPS system by a fast transfer scheme. When the normal and alternate PPS systems are not available, the 6.9-kV PIP buses will be automatically transferred to the standby power supply system. Upon restoration of the normal PPS system, these buses are manually transferred to the normal preferred system. The incoming circuit breakers for the normal and alternate PPS are electrically interlocked to prevent simultaneous connections of UATs and RATs to the PIP buses.

Low-Voltage Alternating Current Power Distribution System

The low-voltage ac power distribution system consists of the onsite electric power distribution circuits that operate at 480 V through 120 V from the power center transformers. The low-voltage ac power distribution system includes power centers, motor control centers (MCCs), distribution transformers, and distribution panels, as well as the associated protective relaying and local instrumentation and control. The power centers are single-fed or double-ended, depending on the redundancy requirements of the loads powered by a given power center. Different buses supply power to the double-ended power center transformers of the PIP buses. Each double-ended power center transformer of the PIP buses will be supplied from different buses, and each will be powered by its normal power supply through its power supply main breaker, with the alternate power supply breaker open. The normal and alternate supply circuit breakers to the power center are electrically interlocked to prevent simultaneous supply to the power center by normal and alternate power supply systems.

Isolation Power Centers

The PIP nonsafety-related buses power the IPCs through step-down transformers, 6.9/0.48 kV, which receive backup power from the standby DGs. The four IPCs, one each for Divisions 1, 2, 3, and 4, are double-ended and can be powered from either of the PIP buses. The normal and alternate source main breakers of each IPC are electrically interlocked to prevent powering the IPC from the normal and alternate sources simultaneously. The IPCs are safety-related and are located in the seismic Category I reactor building in their respective divisional areas. The IPCs supply power to safety-related loads of their respective divisions. These loads consist of the safety-related battery chargers and rectifiers.

The normal and alternate power supply circuits from the PIP buses to the IPC buses are physically separated by distance or physical barriers, so as to minimize, to the extent practical, the likelihood of simultaneous failure under design-basis conditions. The normal power supply circuit-breaker control power, instrumentation, and control circuits are electrically independent and are physically separated from alternate power supply circuit-breaker control power, instrumentation, and control circuits by distance or physical barriers, to minimize, to the extent practical, the likelihood of simultaneous failure under design-basis conditions.

Each IPC will have undervoltage and underfrequency protective relays to protect against degraded voltage and frequency conditions to provide alarms and facilitate IPC bus isolation and transfer functions, using two-out-of-three logic to prevent spurious actuation.

Motor Control Centers

MCCs supply ac power to motors, control power transformers, process heaters, motor-operated valves, and other small electrically operated auxiliaries, including 480 V-to-208 V/120 V and 480 V-to-240 V/120 V transformers. MCCs are assigned to the same load group as the power center that supplies their power.

8.3.1.2.2 Safety-Related Uninterruptible Alternating Current Power Supply System

The safety-related UPS system provides safety-related 120-V ac power to four independent divisions of safety system logic and control, the reactor protection system (RPS), and the safety-related loads requiring uninterruptible power. Each UPS division has two rectifiers, two battery banks, and two inverters. Each rectifier receives 480-V ac power from the IPC of the same division and converts it to 250-V dc power. The 480-V ac/250-V dc rectifier and a safety-related 250-V battery bank, connected through its diodes to a common inverter, convert 250-V dc power to uninterruptible single-phase 120-V ac power. Upon loss of ac power to the IPCs, the safety-related UPS load will be powered automatically by its respective division's safety-related battery through the inverter. The two inverters in each safety-related division will be configured for parallel redundant operation to allow load sharing and the equal discharge of the division's safety-related batteries. Each inverter normally carries approximately 50 percent of the load. If one inverter fails, 100 percent of the load is picked up by the remaining inverter for a period of time greater than 36 hours but less than 72 hours. If both inverters in a division are lost, the associated 120-V ac UPS buses are deenergized. An alarm will sound in the main control room (MCR) for any of the alternate operating lineups.

The plant design and system layout of the UPS provide physical separation of the equipment, cabling, and instrumentation essential to plant safety. The equipment of each division of the safety-related UPS distribution system will be located in an area separated physically from the other divisions. No provisions exist for the interconnection of the safety-related UPS buses of one division with those of another division or nonsafety-related power. All components of safety-related UPS ac systems are housed in seismic Category I structures.

Four divisions of safety-related UPSs provide 120-V ac power for the qualified distributed control and instrumentation system (Q-DCIS) loads/logic components and other safety-related loads requiring uninterruptible power. Two divisions (1 and 2) of safety-related power supply the scram pilot valve solenoids in the RPS, and the same two divisions supply power to the main steam isolation valve solenoids.

Plant staff can conduct routine maintenance on equipment associated with the safety-related UPS system. Inverters, rectifiers, and solid-state switches can be inspected, serviced, and tested channel by channel without tripping the RPS logic.

8.3.1.2.3 Nonsafety-Related Uninterruptible Alternating Current Power Supply System

The nonsafety-related UPS provides uninterruptible ac power for nonsafety-related equipment needed for continuity of plant operation. The five nonsafety-related UPS systems each provide UPS ac power to five load groups (load groups A, B, C, technical support center (TSC)-A, and TSC-B). Two of the UPS systems (A and B) each have two of the following: rectifiers, battery banks, inverters, solid-state transfer switches, manual transfer switches, and regulating transformers. The rectifier and battery bank, connected through the diode to a common inverter, convert to 120-V ac power. The third UPS system (C) has a single set of the power supply, which consists of a battery bank, rectifier, solid-state transfer switch, manual transfer switch, and regulating transformer. Upon loss or failure of the inverter, a static transfer switch automatically transfers nonsafety-related UPS loads from the inverter to a direct ac power supply through the 480-V/120-V regulating transformer.

The normal power supply for each of the nonsafety-related UPS will be through the rectifier and inverter from a nonsafety-related 480-V ac power center. If the 480-V ac power supply fails, transfer from the 480-V ac power supply to the nonsafety-related 250-V dc battery bank occurs automatically. An alarm in the MCR activates when an alternate lineup of the nonsafety-related UPS occurs. The 480-V ac power centers, which provide power to the nonsafety-related battery chargers, are connected to PIP buses that are backed up by onsite standby DGs.

Two dedicated nonsafety-related UPS systems supply ac power to the TSCs in a two-load group configuration. Uninterruptible power for each TSC will normally be supplied from a 480-V ac power center in the same load group. If the normal power supply (through the rectifier and inverter from the 480-V ac power center) fails, a static transfer switch automatically transfers the UPS loads to a direct ac power supply through the regulating transformer. If the 480-V ac power supply fails, a transfer from the 480-V ac power center to the nonsafety-related 125-V dc battery bank, through the inverter, occurs automatically.

8.3.1.2.4 Onsite Standby Alternating Current Power Supply System

The onsite standby ac power system, powered by the two onsite standby DGs, will not be relied on to perform any safety-related function or safe shutdown and, thus, is classified as nonsafety-related. The standby power supply system provides a backup ac power supply to the PIP nonsafety-related buses in the event of a loss of normal and alternate preferred ac power supplies. The PIP buses provide power for various auxiliary and investment protection load groups and safety-related IPCs. An undervoltage relay trips the circuit breaker to the preferred ac power supply and trips major loads on the PIP bus, except for the standby DG auxiliary 480-V power center feeder, before closing the standby ac source breaker. The standby DG starts automatically on loss of bus voltage. When the standby DG reaches full speed and voltage, the standby source breaker will be closed. The large motor loads are connected sequentially and automatically to the PIP bus, after closing the onsite standby ac source breaker.

The source incoming breakers on the PIP buses are interlocked to prevent the inadvertent connection of the onsite standby DG and preferred ac power sources to the PIP buses at the same time. The standby DG, however, can be manually paralleled with the PPS for periodic testing of the generator. Each onsite standby DG operates independently of the remaining standby DG and will be connected to the PIP bus during testing or bus transfer. Each of the onsite standby DGs conforms to the following criteria:

- Capable of starting, accelerating, and supplying its loads in the proper sequence necessary for PIP without exceeding an unacceptable voltage drop

- Capable of reaching full speed and rated voltage within 2 minutes after receiving a signal to start, and being fully loaded within the acceptable time that will not challenge the standby DG capacity

- Has a continuous power rating greater than the sum of the loads of PIP bus and safety-related battery chargers that could be powered concurrently during hot standby, normal plant cooldown, or plant outages

- Has the capability for the generator exciter and voltage regulator system to provide full voltage control during operating conditions, including postulated fault conditions

The standby DG will be shut down and the standby DG breaker will be tripped under the following conditions during all modes of operation and testing:

- Overspeed
- Motoring of generator
- Overload
- Loss of excitation
- Overtemperature
- Ground fault
- Undervoltage
- Overvoltage
- Underfrequency
- Internal fault in generator (differential relay)

The protective functions for these fault conditions of the standby DG and the generator breaker and other off-normal conditions trigger alarms and other indications in the MCR.

Each onsite standby ac power supply can be started or stopped manually from the MCR. Operator action may transfer start/stop control and bus transfer control to a local control station.

The standby ac power supplies are RTNSS systems and their RTNSS functions are discussed in Section 8.4.2.3 of this report.

8.3.1.2.5 Ancillary Alternating Current Diesel Generators

Two nonsafety-related ancillary DGs provide 480V ac power to meet the post-72-hour power requirements following an extended loss of all other ac power sources. The ancillary DGs are RTNSS systems and their RTNSS functions are discussed in Section 8.4.2.3 of this report.

Each ancillary DG output will be connected to a 480-V ancillary diesel bus. The ancillary DGs are seismic Category II, as are their associated auxiliaries, controls, electrical distribution buses, and fuel oil tanks. The ancillary DGs and associated equipment are housed in a seismic Category II structure. The ancillary power will not be required to support safety-related loads for the first 72 hours following the loss of all other ac power sources.

The ancillary DGs also have the capability to support prestart and starting functions for the onsite standby DGs, if they failed to start upon initial demand and required a delayed start. Power can also be supplied to the nonsafety-related 125-V dc battery chargers, with their batteries disconnected, to power equipment such as the protective relaying and breaker controls required for restoring offsite or onsite standby DG power to the ESBWR systems and equipment.

These 480-V ancillary diesel buses are normally powered by the offsite power supply systems or onsite standby DGs through the PIP buses and the 6900/480-V step-down transformers. On sensing undervoltage to their buses or a low ancillary diesel room temperature, the 480-V ac ancillary diesel bus feeder breaker will trip and send a start signal to the ancillary DG. The signal starts the ancillary DG and closes the ancillary generator power supply breaker.

8.3.1.2.6 Electric Heat Tracing

The electric heat tracing system provides freeze protection, where required, for outdoor service components and for warming process fluids, if required, either indoors or outdoors. The heat tracing will be safety-related and will be powered from safety-related distribution buses, if heat tracing is required for proper operation of a safety-related system. Safety-related heat tracing will be assigned to the appropriate division of safety-related power. Nonsafety-related heat tracing will be supplied from the same power center or MCC as the components protected.

8.3.1.2.7 Cathodic Protection

A cathodic protection system will be provided to the extent required. Its design will be plant–specific, be tailored to the site conditions, and meet the requirements of the National Association of Corrosion Engineers standards. COL Information Item 8A.2.3-1-A states that the COL applicant will provide the minimum requirements for the design of the cathodic protection system.

8.3.1.2.8 Safety-Related Systems Description

1. Physical Separation and Independence

    - Electrical equipment will be separated in accordance with RG 1.75, GDC 17, and IEEE Std 384.

    - Separation and independence of safety-related equipment will be achieved by using separate safety-related structures, barriers, or a combination thereof.

    - To meet the provisions of policy issue SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," issued January 1989, which relates to fire tolerance, the design calls for 3-hour-rated fire barriers among areas of different safety-related divisions throughout the plant, except in the primary containment and the control room complex.

    - The safety-related electrical equipment will be located in separate seismic Category I rooms in the reactor building to ensure electrical and physical separation among the divisions.

    - Electrical and control equipment, panels and racks, and cables and raceways grouped into separate divisions are identified by color-coding, so that their electrical divisional assignment will be apparent and so that an observer can visually differentiate between safety-related equipment and wiring of different divisions, and between safety-related and nonsafety-related equipment and wiring.

    - Independence of the electrical equipment and raceway systems, among the different divisions, will be maintained primarily by a firewall-type separation, where feasible, and by spatial separation.

2. Design Bases and Criteria

    - Plant design specifications for electrical equipment require that such equipment be capable of continuous operation with equipment terminal voltage fluctuations of plus or minus (±) 10 percent of rated voltage limits.

- Power supply systems will be capable of supplying the power of voltage and frequency within acceptable tolerances.

- The interrupting capacity of distribution panels will be at least equal to the maximum available fault current to which the panels are exposed under all modes of operation. Circuit breakers and their applications are in accordance with American Nuclear Standards Institute specification, ANSI C37.50-1989, “1989 Switchgear – Low-Voltage AC Power Circuit Breakers Used in Enclosures – Test Procedures”.

- Refurbished circuit breakers shall not be used in either safety-related or nonsafety-related circuitry in the ESBWR design. New circuit breakers shall be specified in all ESBWR purchase specifications. (NRC Bulletin (BL) 88-10, “Nonconforming Molded-Case Circuit Breakers,” issued November 1988, and Information Notice 88-46, “Licensee Report of Defective Refurbished Circuit Breakers,” issued July 1988, identify problems with defective refurbished circuit breakers.)

3. Testing

   The design provides for periodic testing of the channel from the sensing devices through actuated equipment to ensure that safety-related equipment will function in accordance with design requirements and the requirements of RG 1.118 and IEEE Std 338.

8.3.1.2.9 Electrical Circuit Protection Description

Protective relays will be used to isolate a fault. Protective relay schemes and direct acting trip devices will be provided throughout the onsite power system to do the following:

- Isolate faulted equipment, circuits, or both from the power system
- Prevent damage to equipment
- Protect personnel
- Minimize system disturbances
- Maintain continuity of the power supply

1. Grounding

   - The electrical grounding system will comply with the guidelines provided in IEEE Std 665-1995 and IEEE Std 1050-1996. The electrical grounding system comprises the following:

     – Instrument and computer grounding network

     – Equipment grounding

     – Plant grounding grid

     – Lightning protection network for protection of transformer and equipment located outside buildings

   The plant instrumentation will be grounded through a separately insulated radial grounding system composed of buses and insulated cables. The instrumentation grounding system will be connected to a discrete point of the station grounding grid at a dedicated instrumentation grounding rod by exothermic welding. The instrumentation grounding system will be insulated from all other grounding and surge protection circuits, up to the point of connection at the ground grid. A separate instrumentation grounding system will be provided for plant analog and digital instrumentation systems.

The equipment grounding network will be such that all major equipment, structures, and tanks are grounded with two diagonally opposite ground connections. The ground bus of all switchgear assemblies, MCCs, and control cabinets will be connected to the station ground grid through at least two parallel paths. Bare copper risers will be furnished for all underground electrical ducts and equipment and for connections to grounding systems within buildings. One bare copper cable will be installed with each underground electrical duct run, and all metallic hardware in each manhole will be connected to the cable.

A plant grounding grid, consisting of bare copper cables, limits step and touch potentials to safe values under all fault conditions. The buried grid will be connected to the ground mat at the switchyard and connected to systems within the buildings by a bare copper loop, which encircles each building.

The plant's main generator will be grounded with a neutral grounding device to limit the magnitude of fault current resulting from a phase-to-phase fault. Although the impedance of the neutral grounding device limits the maximum phase current under short-circuit conditions, it does not limit the current to a value less than that for a three-phase fault at its terminals.

The onsite, medium-voltage ac distribution system will be resistance-grounded at the neutral point of the low-voltage windings of the UATs and RATs. The neutral point of the generator windings of the onsite standby ac power supply will be through neutral grounding resistors, sized for continuous operation in the event of a ground fault.

The neutral point of the low-voltage ac distribution systems will be either solidly or impedance grounded to ensure proper coordination of ground fault protection. The dc systems are ungrounded.

2. Lightning Protection

The lightning protection system covers all major plant structures and will be designed to prevent direct lightning strikes to the buildings, electric power equipment, and instruments. It consists of air terminals, bare downcomers, and buried grounding electrodes. Lightning arresters will be provided for each phase of all tie lines connecting the plant electrical systems to the switchyard and offsite lines. These arresters will be connected to the high-voltage terminals of the main step-up transformers, UATs, and RATs. Plant instrumentation located outdoors or connected to cabling running outdoors has surge suppression devices to protect the equipment from lightning-induced surges.

3. Bus Protection

The incoming circuit breakers to the medium-voltage (13.8-kV and 6.9-kV) bus will be equipped with inverse-time overload, ground fault, bus differential, undervoltage, and degraded voltage protection.

Feeder breakers for power centers and the medium-voltage motors will be equipped with instantaneous, inverse-time overload, and ground fault protection.

Feeder breakers for 480-V MCC buses will be equipped with long-time and short-time overload and ground fault protection.

The IPC buses will be equipped with inverse-time overload and ground fault protection. In addition, loss of voltage, degraded voltage, and underfrequency protection serve to isolate these buses from the nonsafety-related system under degraded conditions of voltage and frequency.

The 480-V MCCs will be equipped with instantaneous and inverse-time overload protection. The 480-V power center motor feeder breakers have instantaneous, inverse-time overload and ground fault protection.

4. Containment Electric Penetrations

   Separate electrical penetrations are provided for circuits of each safety-related division and for nonsafety-related circuits. The circuits of each electrical penetration are of the same voltage class. Redundant overcurrent interrupting devices are provided for electrical circuits routed through containment penetrations, if the maximum available fault current will be greater than the continuous rating of the penetration. This avoids penetration damage in the event of failure of any single overcurrent device to clear a fault within the penetration or beyond it. Electrical penetration assemblies of different safety-related divisions are separated by 3-hour-rated fire barriers, are in separate rooms, or are located on separate floor levels. Separation by distance without barriers will be allowed only in the inerted containment. Separation between divisional and nondivisional penetrations will be in accordance with IEEE Std 384. Grouping of circuits in penetration assemblies follows the same raceway voltage groupings. Electrical penetrations design bases include compliance with GDC 50 and following the guidance of RG 1.63.

8.3.1.2.10 Load Shedding and Sequencing on Plant Investment Protection Buses Description

Load shedding, bus transfer, and sequencing on the 6.9-kV PIP buses are initiated on loss of bus voltage. Only loss of preferred power (LOPP) to the 6.9-kV PIP bus trips the loads on the bus. The standby DG protective relaying (voltage and frequency) logic and control system for the electric power distribution system generates PIP bus ready-to-load signals.

Onsite standby DGs are of sufficient size to accommodate required loads with an acceptable starting sequence.

- LOPP

  The 6.9-kV PIP buses are normally energized from the normal PPS. When the normal PPS system is not available, a fast transfer scheme will be activated to transfer power from the normal preferred supply to the alternate preferred supply. If both PPS systems fail, incoming circuit breakers to the PIP bus will trip, and the loads on the bus will shed through an undervoltage signal. The signal starts the onsite standby DGs and closes the standby power supply breaker with an acceptable level of voltage and frequency. The loads will be started in sequence, as required. Transfer back to the PPS will be a synchronized closure of the feeder breaker by manual action.

- LOCA

  A LOCA that occurs without a LOPP has no effect on the onsite ac electrical distribution system. The plant remains on a preferred power source, and the onsite standby diesel generator will not be started.

- LOPP Following LOCA

  If the bus voltage (normal or alternate preferred power) is lost during post-accident operation, transfer to the standby onsite ac power source occurs as described in the LOPP section above.

- LOCA Following LOPP

  If a LOCA occurs following the loss of both the normal and alternate preferred power supplies, the standby onsite ac power source should have already started from the low bus voltage. As discussed in the LOPP section above, automatic load sequencing will start.

- LOCA when Onsite Standby Is Parallel to Preferred Power Supply System during Testing

  If a LOCA occurs when the standby DG is paralleled with either of the PPS systems through the 6.9-kV PIP bus, the standby DG automatically disconnects from the 6.9-kV PIP bus.

- Loss of Normal PPS during Onsite Standby Power Supply System Paralleling Test

  If the normal PPS is lost during the standby onsite ac power source paralleling test, the normal PPS breaker and standby DG breaker are automatically tripped and the alternate PPS accepts loads to reenergize the selected bus loads.

- Loss of Alternate PPS during Onsite Standby Power Supply System Paralleling Test

  If the alternate PPS is used for load testing the standby DG and the alternate PPS is lost, the alternate PPS breaker and standby DG breaker are automatically tripped. The affected bus will then be transferred back to the normal PPS.

8.3.1.2.11 Raceway and Cable Installation

Power and control cables are specified for continuous operation at conductor temperatures not exceeding 90 degrees Celsius (C) (194 degrees Fahrenheit (F)) and should withstand an emergency overload temperature of up to 130 degrees C (266 degrees F), in accordance with Insulated Cable Engineers Association (ICEA) S 95-658/ National Electrical Manufacturers Association (NEMA) WC-70, “Non-shielded 0-2 kV Cables.” The base ampacity rating of the cables will be established as published in IEEE Std 835, “IEEE Standard Power Cable Ampacity Tables,” and ICEA P-54-440/NEMA WC-51, “Ampacities of Cable in Open-Top Cable Trays.”

Cables are specified to continue to operate at 100 percent relative humidity with a service life expectancy of 60 years. Safety-related cables are designed to survive the LOCA ambient condition at the end of the 60-year lifespan. Certified proof tests are performed on cables to demonstrate a 60-year lifespan, and resistance to radiation, flame, and the environment. The testing methodology ensures that such attributes are acceptable for the 60-year lifespan.

All cables specified for safety-related systems and circuits are moisture- and radiation-resistant, are highly flame-resistant, and evidence little corrosive effect when subjected to heat or flame, or both. Certified proof tests are performed on cable samples.

Cable tray fill will be limited to 40 percent of the cross-sectional area for trays containing power cables and 50 percent of the cross-sectional area for trays containing instrumentation and control cables. If tray fill exceeds the above maximum fills, the tray fills are justified and documented. Medium-voltage cable tray fill will be a single layer with maintained spacing.

Cable splices in the raceway are prohibited. Cable splices are made only in manholes, boxes, or suitable fittings. Splices in cables passing through the containment penetration assemblies are made in terminal boxes located adjacent to the penetration assembly.

Three-hour fire rated concrete barriers are used among the RATs, the UATs, and the main transformers and spare main transformer, including the containment and collection of transformer oil. The concrete barriers provide separation and independence of the system.

Cables are installed in trays in accordance with their voltage ratings. The raceways are arranged, physically, top to bottom, based on the function and the voltage class of the cables. Each division of safety-related ac and dc system cables will have its own independent and separate raceway system.

8.3.1.3 *Staff Evaluation*

Industry experience has shown that the voltage transient during islanding (the main generator supplying station auxiliary loads with offsite power disconnected) after loss of the electrical grid, or a generator voltage regulator malfunction, can propagate through the plant's electrical distribution system, resulting in tripping or a loss of safety-related equipment. In RAI 8.2-14, the staff asked how the ESBWR design accommodates the voltage and frequency transient. In response to RAI 8.2-14, the applicant stated that the plant's safety-related and nonsafety-related equipment is designed to accommodate operational voltage and frequency transients, such as the islanding transient, and faulted conditions, such as generator voltage regulator malfunctions. The Forsmark incident, the subject of NRC Information Notice 2006-18, "Significant Loss of Safety-Related Electrical Power at Forsmark, Unit 1, in Sweden," highlights the importance of coordinating the protective trip among the UPS input rectifiers, battery chargers, and inverters. The ESBWR battery chargers and UPS input rectifiers are designed to accommodate the expected islanding transient without tripping. The design includes trip coordination among input rectifiers, battery chargers, and inverters, so that rectifiers and battery chargers trip on excessive high bus voltage and inverters continue to supply the safety-related loads, using stored energy from safety-related batteries. The protective relaying scheme for the ESBWR will be completed as part of the detailed design, to ensure protection for the safety-related and nonsafety-related equipment, as required. However, the staff determined that additional details on the UPS (rectifier and inverter) and battery charger (rectifier) response to high bus voltage were needed.

In RAI 8.2-14 S01, the staff asked the applicant how the rectifier and inverter trip are coordinated. Additionally, the staff asked the applicant to explain the impact of excessive high bus voltage on safety-related loads when fed from the regulating transformer during the islanding mode of operation. In its response, the applicant stated that the safety-related battery chargers and UPS input rectifier high dc voltage trips are coordinated, such that the associated inverters do not trip on high dc input voltage during voltage transients on the ac distribution system. The trips are coordinated such that the inverter high dc input voltage trip setpoint will be greater than the associated battery charger and UPS input rectifier high dc output trip setpoints. In addition, the time delay for the inverter high dc input voltage trip will be greater than the time delay for the battery charger and UPS input rectifier high dc output voltage trips.

In this way, the high dc voltage protection will be coordinated in both magnitude and time, so the battery charger and UPS input rectifier always trip before their dc output voltage reaches the level that would cause an inverter trip on high dc input voltage. This is a functional requirement for the battery charger and UPS equipment; the actual trip magnitude and time margins are a function of the vendor-specific equipment design.

Additionally, the applicant stated that it eliminated the safety-related UPS bypass transformers from the ESBWR design because of the potential for disruptive voltages and frequencies to reach the safety-related loads. The 100-percent redundant UPS rectifiers and inverters within each division negate the need for a bypass transformer. The battery chargers and UPS will be designed to have the required fault clearing and load inrush capability, without the need to switch to a bypass source. Each inverter within a division can be taken out of service for maintenance without the need to deenergize the UPS bus; however, the division will be inoperable according to the technical specifications (TSs). The applicant revised the DCD Tier 1 and Tier 2, Revision 5, to reflect the removal of the safety-related UPS bypass transformer and other changes discussed above. Based on the applicant's response, RAI 8.2-14 is resolved. The staff confirmed that Revision 6 of the DCD includes the changes described above.

In RAI 14.3-413, the staff asked the applicant to provide an ITAAC to verify the trip coordination of safety-related battery chargers and UPS input rectifiers with inverters. In its response, the applicant revised DCD Tier 1, Revision 5, to include the requirement to verify the trip coordination of the safety-related battery chargers and UPS input rectifiers with inverters. This new DCD Tier 1 ITAAC is based on new information added to DCD Tier 2 that discusses coordination of the rectifier and inverter high dc voltage trips. Based on the applicant's response, RAI 14.3-413 is resolved. Section 14.3.6 of this report discusses RAI 14.3-413 regarding the ITAAC. The staff confirmed that Revision 6 of the DCD includes the changes described above.

In RAI 14.3-394 S01, the staff asked the applicant to provide an interface requirement for demonstrating the capacity and capability of the offsite power system. In its response, the applicant incorporated the PPS definition into DCD Tier 2, Revision 6, Chapter 8, per IEEE Std 765. The PPS consists of the normal and alternate PPSs and includes those portions of the offsite and onsite power systems required for power flow from the offsite transmission system to the safety-related IPC incoming line breakers. Additionally, the applicant revised DCD Tier 2, Revision 5, Chapter 8, to clarify that GDC 17 applies to the entire PPS and added ITAAC in Tier 1, Section 2.13.1, to address capacity, capability, and the physical and electrical separation of the normal and alternate PPS. Based on the applicant's response, RAI 14.3-394 S01 is resolved. Section 14.3.6 of this report discusses RAI 14.3-394 S01 regarding ITAAC. The staff confirmed that Revision 6 of the DCD includes the changes described above.

In RAI 8.2-16, the staff asked the applicant to discuss the protection of the main transformer, UATs, RATs, isolated phase bus, and nonsegregated phase bus. Additionally, the staff asked the applicant to provide the rating of these devices. In its response, the applicant modified DCD Tier 2, Revision 5, Section 8.3.1.1, to add that the main transformer, UATs, and RATs are protected against overcurrent, differential current, ground overcurrent, and sudden overpressure. Additionally, the applicant modified Section 8.3.1.1 to include the protection of the isolated phase bus and nonsegregated bus against overcurrent and bus differential. The ratings of the devices will not be known until the completion of the ESBWR detailed design. However, the response to RAI 14.3-394 S01 added ITAAC Item 9 to DCD Tier 1, Revision 6, Section 2.13.1, to ensure the equipment within the onsite portion of the PPS is rated to supply necessary load requirements, including power, voltage, and frequency, during design-basis operating modes. This ITAAC confirms that the transformer and buses are sized and rated appropriately. Based on the applicant's response, RAI 8.2-16 is resolved. The staff confirmed that Revision 6 of the DCD includes the changes described above.

In RAI 8.3-60, the staff asked the applicant to clarify several issues concerning degraded voltage conditions. In response, the applicant stated that IPC buses are double-ended, such that each bus can receive power from either standby DG-backed 6.9-kV PIP bus. The IPC buses are equipped with degraded voltage and frequency protection to facilitate bus isolation and transfers. This relaying also serves as a backup to the ac input monitoring built into the safety-related battery chargers and UPS for protection against voltage and frequency transients on the ac supply. Additionally, the applicant revised DCD Tier 2, Revision 5, Section 8.3.1.1.2, to state that IPCs are protected against degraded voltage and frequency conditions by using voltage and frequency relays installed in each IPC to provide alarms and facilitate IPC bus isolation and transfer functions, using two-out-of three logic to prevent spurious actuation. The applicant further stated that degraded voltage and undervoltage protection is provided on the medium-voltage bus incoming line breakers. This provides protection for all medium-voltage buses, low-voltage buses, and connected loads from degraded voltage conditions on the normal and alternate preferred offsite power sources. Underfrequency protection will not be provided on medium-voltage or low-voltage buses. Plant auxiliary loads are able to operate within a ± 5 percent frequency tolerance; the grid underfrequency load shedding scheme will act to maintain the frequency well within this band, or turbine generators on the line will begin to trip within a few seconds. If islanding or connected to the onsite standby DGs, underfrequency relaying associated with these sources will actuate and remove the out-of-tolerance source from the bus. The safety-related frequency protection on the IPC bus will remove it from the source to protect the safety-related loads.

The staff finds that the safety-related loads fed from the IPC bus are battery chargers (rectifiers) and UPSs. The safety-related loads are able to operate within ±10 percent voltage tolerance. The UPS consists of a rectifier and an inverter. Each rectifier receives 480-V ac power from the IPC bus and converts it to 250-V dc power. The 250-V dc is then inverted to 120-V ac (uninterruptible) by the inverters. The degraded voltage scheme will initiate transfer to the alternate PPS (i.e., RAT) if the normal PPS is degraded (i.e., UAT problem). Also, the degraded voltage protection on the isolation bus will remove it from the source if both normal preferred and alternate preferred power supplies are degraded. Under this situation, the inverters continue to supply safety-related loads (e.g., Q-DCIS, scram pilot valves, main steam isolation valve solenoids), using stored energy from the safety-related batteries. Based on the above, the staff finds that the applicant has adequately addressed degraded voltage issues and, therefore, RAI 8.3-60 is resolved. The staff confirmed that Revision 6 of the DCD includes the changes described above.

In RAI 14.3-427, the staff asked the applicant to provide ITAAC for grounding and lightning protection systems. In its response, the applicant added ITAAC, as the staff requested. The applicant also removed from DCD Tier 2, Revision 5, Appendix 8A.1.1, the statement that lightning protection ground rods would be separate from the normal grounding system. The ITAAC verifies that a connection exists between the lightning protection system and the station ground grid. The applicant stated that this change, and allowing the lightning protection ground rods to tie to the ground grid, will make the lightning protection system more robust by providing additional volume to adequately dissipate lightning strikes. The staff finds that the applicant adequately addressed the issue and RAI 14.3-427 is resolved. Section 14.3.6 of this report discusses RAI 14.3-427 regarding ITAAC. The staff confirmed that Revision 6 of the DCD includes the changes described above.

DCD Tier 2, Revision 6, Section 8.3.3.2, did not address the use of underground cables or cables in a wetted environment. Operating experience has shown that cross-linked polyethylene or high-molecular-weight polyethylene insulation materials are susceptible to water tree formation. Cable failures have a variety of causes: manufacturing defects, damage caused by shipping and installation, and exposure to electrical transients or abnormal environmental conditions during operation. Electrical cables in nuclear power plants are usually located in dry environments, but some cables are exposed to moisture from condensation and wetting in inaccessible locations, such as buried conduits, cable trenches, cable troughs, aboveground and underground duct banks, underground vaults, and direct buried installations. Since underground cables are susceptible to moisture, in RAI 8.3-67, the staff asked the applicant to identify the cables that are inaccessible or routed underground that support equipment and other systems within the scope of 10 CFR 50.65 (the maintenance rule). Additionally, the staff asked the applicant to indicate whether there are any plans to implement a program for testing and inspection of inaccessible or underground power, control, and instrumentation cables, in accordance with Generic Letter (GL) 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," issued February 2007; and the frequency for such testing and inspection, or to provide justification for not developing such a program. In its response, the applicant stated that the standard ESBWR plant design will have its principal plant structures, as listed in DCD Tier 2, Revision 9, Section 1.2.1, connected by a series of tunnel structures that will enable the routing of cables and raceways in areas not subject to water intrusion. For other ESBWR structures not listed in Section 1.2.1 as well as site specific structures requiring electric power, the detail design is within owner's yard scope, not considered part of the standard plant design and is covered by the COL applicants. The applicant identified two ESBWR systems (plant service water system [PSWS] and standby DG fuel oil transfer system) with accident mitigating functions that have power and control cables that are in a potentially wetted environment due to manholes. These two systems are in the COL applicant's yard scope and covered by the COL applicant per the maintenance rule. The applicant revised DCD Tier 2, Revision 6, Subsection 8.3.3.2 and Table 1.10-1 to add a new COL Information Item 8.3.4-2-A. The COL Information Item 8.3.4-2-A requires the identification and monitoring of underground or inaccessible power and control cables to the PSWS and standby DG fuel oil transfer system equipment that have accident mitigating functions. Additionally, the applicant revised DCD Tier 2, Revision 6, Subsection 8.3.3.2 to include the statement, "A water-tree formation retardant is specified when polyethylene cable insulation is selected for medium voltage use. A dry cure process is specified for cable insulation." The staff finds the applicant's addition of COL Information Item 8.3.4-2-A is appropriate to address the issue discussed in GL 2007-01 and COL Information Item 8.3.4-2-A is consistent with the guidance of SRP Section 8.3.1. Additionally, the staff finds that the selection of water-tree formation retardant and dry cure process for cable insulation for medium voltage cable will improve medium voltage cable design for wetted environment. Based on the applicant's response, RAI 8.3-67 is resolved. The staff confirmed that Revision 7 of the DCD includes the changes described above.

The following paragraphs analyze compliance with the GDC and consistency with the RGs and other SRP guidance:

- GDC 2 and GDC 4

  All components of the safety-related IPCs and UPS system are housed in seismic Category I structures designed to protect them from natural phenomena. These components are qualified to the appropriate seismic, hydrodynamic, and environmental conditions as part of the qualification program described in DCD Tier 2, Revision 9, Section 3.11 and evaluated in Section 3.11 of this report. The safety-related IPCs and UPS system are in compliance with the requirements of GDC 2 and 4.

- GDC 5, RG 1.81, and RG 1.32

  The ESBWR plant is designed as a single-unit plant, and, thus GDC 5, RG 1.81, and RG 1.32 are not applicable.

- GDC 17

  The safety-related dc power supply supports passive core cooling and containment safety-related functions. The ESBWR design complies with GDC 17 with respect to two independent and separate offsite power supply and onsite standby power supply systems.

- RG 1.6

  The standby power sources (i.e. standby DGs) are nonsafety-related. The standby power sources are designed with the required independence. The 120-V UPS systems, together with safety-related IPCs, are designed with the required independence. The design is consistent with the guidance of RG 1.6.

- RG 1.9

  The ESBWR design does not require safety-related DGs, and hence, RG 1.9 is not applicable.

- RG 1.32

  The design provides for safety-related 120-V UPS systems to support passive core cooling and containment integrity safety functions. The design is consistent with the guidance of RG 1.32.

- RG 1.53

  The safety-related 120-V UPS system is designed so that no single active failure in any division of the 120-V UPS system results in conditions that could prevent the safe shutdown of the plant while a separate division is out of service for maintenance. Therefore, the design is consistent with the guidance of RG 1.53.

- RG 1.75

  The ESBWR design provides the physical separation and independence of the division of the electrical circuit and equipment comprised of, or associated with, the Class 1E power systems, Class 1E protection systems, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and systems it actuates to perform their safety-related functions. The design provides separation to maintain the independence of sufficient circuits and equipment so that the protective functions required during and following any design-basis event can be accomplished. Also, this design provides physical and electrical separation of safety-related circuits from nonsafety-related circuits. Therefore, the ESBWR design is consistent with the recommendations of RG 1.75, which is related to GDC 17.

- RG 1.153

  Safe shutdown relies upon the 120-V UPS system and meets the design requirements for redundancy. The redundant safety-related loads are distributed between redundant distribution systems, and power systems are supplied from the related redundant distribution systems. Therefore, the design is consistent with the guidance of RG 1.153.

- RG 1.155

  The ESBWR design does not require safety-related emergency DGs, and hence, this RG is not applicable.

- RG 1.204

  The lightning arresters are connected to the high-voltage terminals of the main step-up transformers, UATs, and RATs. Plant instrumentation located outdoors or connected to cables running outdoors has surge suppression devices to protect the equipment from lightning-induced surges. The design is consistent with the guidance of RG 1.204.

- GDC 18

  The offsite and onsite power systems that supply ac power to SSCs important to safety are testable. Thus, the ESBWR design complies with GDC 18 and is consistent with the guidance of RG 1.32, RG 1.47, RG 1.118, and RG 1.153.

- GDC 33, 34, 35, 38, 41, and 44

  The potential risk contribution of a design-basis event is minimized because the passive reactor design does not require ac power sources for such events. Passive reactor designs incorporate passive safety-related systems for core cooling and containment integrity and, therefore, do not depend on the onsite standby power source. They are designed to automatically establish and maintain safe shutdown conditions after design-basis events for 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. Therefore, the ESBWR design is not required to meet the requirements of GDC 33, 34, 35, 38, 41, and 44.

- GDC 50

  Redundant overcurrent interrupting devices are provided for electrical circuits routed through containment penetrations if the maximum available fault current is greater than the continuous rating of the penetration. This avoids penetration damage in the event of a failure of any single overcurrent device to clear a fault within the penetration or beyond it. Electrical penetrations are in compliance with GDC 50 and follow the guidance of RG 1.63.

- 10 CFR 50.63

  With regard to 10 CFR 50.63, the ESBWR design bases do not rely on an onsite ac power system to achieve and maintain safe shutdown (see the SBO evaluation in Sections 8.4.2.1 and 15.5.5 of this report). RG 1.9 and RG 1.155, regarding a reliability program for emergency onsite ac power source, are not applicable.

- 10 CFR 50.65(a) (4), RG 1.160, and RG 1.182

  As described in DCD Tier 2, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

The applicant has provided information to demonstrate that the ancillary DG will be capable of providing the post-72-hour power requirements to onsite loads, including the safety-related UPS loads. These and other RTNSS functions of the ancillary DG are discussed in Section 8.4.2.3 of this report. Also, this system design provides the isolation devices to separate the safety-related and nonsafety-related systems. Therefore, the staff finds the ancillary ac DGs to be acceptable.

8.3.1.4 *Combined License Unit-Specific Information*

The applicant stated that the COL applicant will address the cathodic protection system via COL Information Item 8A.2.3-1-A. Additionally, the COL applicant will address the underground or inaccessible power and control cable monitoring program via COL information item 8.3.4-2-A.

These COL items identify two items related to the ac power system that the COL applicant will address. The staff agrees that these items are site-specific and therefore appropriately addressed by the COL applicant. In addition, these COL items for the ac power system, provided they are adequately addressed, provide reasonable assurance that the ac power system meets the requirements of GDC 17. Therefore, the staff finds this COL information item acceptable.

8.3.1.5 *Conclusion*

The staff has reviewed the onsite ac power supply system, including the UPS systems. Based on its review, the staff concludes that the applicant has provided sufficient information to demonstrate that the onsite ac power supply systems are consistent with the guidance of cited RGs and are capable of providing the power supply to onsite loads needed to support the plant’s safe operation. The design of the onsite ac power supply systems is acceptable and meets the requirements of GDC 2, 4, 17, 18, and 50.

8.3.2 Direct Current Power Systems

8.3.2.1 *Regulatory Criteria*

The dc power systems include those dc power sources (and their distribution systems and auxiliary supporting systems) that supply motive or control power to safety-related and nonsafety-related equipment. The staff’s review covers the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, 44, and 50, and on 10 CFR 50.63, 10 CFR 50.55a(h), and 10 CFR 50.65(a)(4), as they relate to the capability of the onsite electrical power system to facilitate the functioning of SSCs important to safety. SRP Sections 8.1 and 8.3.2 contain specific review criteria.

Acceptance criteria for the evaluation of dc power systems (onsite) are based on meeting the following relevant requirements:

- GDC 2, as it relates to the ability of dc power system SSCs to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapter 3 of this report and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering

- GDC 4, as it relates to the ability of dc power system SSCs to withstand the effects of missiles and environmental conditions associated with normal operation and postulated accidents, as established in Chapter 3 of this report and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering

- GDC 5, as it relates to sharing dc power system SSCs

- GDC 17, as it relates to (1) the capacity and capability of the onsite dc power system to enable the functioning of SSCs important to safety, and (2) the independence and redundancy of the onsite dc power system in performing its safety function, assuming a single failure

- GDC 18, as it relates to the testability of the onsite dc power system

- GDC 33, 34, 35, 38, 41, and 44, as they relate to the operation of the onsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in GDC 33, 34, 35, 38, 41, and 44 are accomplished

- GDC 50, as it relates to the design of containment electrical penetrations containing circuits of safety-related and nonsafety-related dc power systems

- 10 CFR 50.63, as it relates to the ability of the onsite dc power system to support the plant in withstanding, or coping with and recovering from, an SBO event

- 10 CFR 50.55a (h), as it relates to the incorporation of IEEE Std 603-1991 (including the correction sheet, dated January 30, 1995) and IEEE Std 279 for protection and safety systems

- 10 CFR 50.65(a) (4), as it relates to the assessment and management, before the performance of maintenance activities, of the increase in risk that may result from proposed maintenance activities, including but not limited to, surveillances, post-maintenance testing, and corrective and preventive maintenance (noting that SRP Chapter 17 reviews compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein)

SRP Section 8.3.2 provides guidance on how an application can meet the above regulations.

- RG 1.6, Regulatory Positions D.1, D.3, and D.4, as they relate to the independence between redundant onsite dc power sources and between their distribution systems

- RG 1.32, as it relates to the design, operation, and testing of the safety-related portions of the onsite dc power system, noting that, except for sharing safety-related dc power systems in multiunit nuclear power plants, RG 1.32 endorses IEEE Std 308-2001

- RG 1.75, as it relates to the physical independence of the circuits and electrical equipment that comprise or are associated with the onsite dc power system

- RG 1.81, as it relates to the sharing of SSCs of the dc power system, noting that Regulatory Position C.1 states that multiunit sites should not share dc systems

- RG 1.128, "Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants," issued October 1978, as it relates to the installation of vented lead-acid (VLA) storage batteries in the onsite dc power system

- RG 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," issued February 1978, as it relates to maintenance, testing, and replacement of vented lead-acid (VLA) storage batteries in the onsite dc power system

- RG 1.118, as it relates to the capability to periodically test the onsite dc power system

- RG 1.153, as it relates to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems of nuclear plants, including the application of the single-failure criterion in the onsite dc power system, noting that, as endorsed by RG 1.153, IEEE Std 603 provides a method acceptable to the staff to evaluate all aspects of the electrical portions of the safety-related systems, including basic criteria for addressing single failures

- RG 1.53, as it relates to the application of the single-failure criterion

- RG 1.63, as it relates to the capability of electric penetration assemblies in containment structures to withstand a LOCA without loss of mechanical integrity and the external circuit protection for such penetrations

- RG 1.155, as it relates to the capability and the capacity of the onsite dc power system to withstand an SBO, including batteries associated with the operation of the alternate ac power source(s) (if used)

- RG 1.160, as it relates to the effectiveness of maintenance activities for dc power systems

- The guidelines of RG 1.182, as they relate to conformance to the requirements of 10 CFR 50.65(a) (4) for assessing and managing risk when performing maintenance

8.3.2.2 *Summary of Technical Information*

The onsite dc power systems consist of safety-related and nonsafety-related power systems. Each system consists of an ungrounded battery bank, battery chargers, and dc distribution equipment.

The design provides for eight independent safety-related Class 1E 250-V dc batteries, two each for Divisions 1, 2, 3, and 4. They provide four divisions of independent and redundant onsite dc power supplies for safety-related loads, monitoring, and emergency lighting for the MCR and the remote shutdown area.

The design provides for seven independent nonsafety-related dc batteries, consisting of five 250-V dc and two 125-V dc batteries. The nonsafety-related dc systems supply dc power for control and switching, switchgear control, TSC, instrumentation, and station auxiliaries.

The Class 1E dc system also supplies power for the safe shutdown of the plant without the support of battery chargers, during a loss of all ac power sources coincident with a DBA. The system will be designed so that no single failure will result in a condition that will prevent the safe shutdown of the plant.

The non-Class 1E dc system provides power to the plant's non-Class 1E control and instrumentation equipment and loads that are required for plant operation and investment protection. Operation of the non-Class 1E dc supply system will not be required for plant safety.

8.3.2.2.1 Safety-Related Direct Current System

Safety-related Divisions 1, 2, 3, and 4 each consists of two separate 250-V dc battery banks. Each battery bank supplies dc power to the loads through the safety-related inverter for at least 72 hours following a design-basis event. Each of the safety-related battery systems has a 250-V battery bank, battery charger, main distribution panel, and ground detection panel. One divisional battery charger will be used to supply each group dc distribution panel and its associated battery. The divisional battery charger will be fed from its divisional 480-V isolation power center.

Each division has a standby charger as backup to either of the battery banks of its respective division.

The 250-V dc systems supply dc power to Divisions 1, 2, 3, and 4 and are designed as safety-related equipment, in accordance with IEEE Std 308 and IEEE Std 946, "IEEE Recommended Practice for the Design of Safety-Related dc Auxiliary Power Systems for Nuclear Power Generating Stations." The design ensures that no single active failure in any division of the system results in conditions that prevent the safe shutdown of the plant while a separate division is out of service for maintenance.

The plant design and circuit layout of the dc systems provide physical separation of equipment, cabling, and instrumentation essential to plant safety. Each 250-V dc battery will be separately housed in a ventilated room apart from its charger, distribution, and ground detection panels. Equipment of each division of the dc distribution system will be located in an area separated physically from the other divisions. All components of safety-related 250-V dc systems are housed in seismic Category I structures.

Each division has two 250-V safety-related batteries, and each battery supplies power to its safety-related inverter for at least 72 hours following a DBA. The minimum dc system battery bank terminal voltage at the end of the discharge period will be 210 V (1.75 V per cell). The maximum equalizing charge voltage for safety-related batteries will be site-specific and specified by the battery vendor and as allowed by the voltage rating of the connected loads (inverters). The UPS inverters are designed to supply 120-V ac power with dc input less than the minimum discharged voltage of 210-V dc and greater than the maximum equalizing charge voltage, which is site specific, as specified by the battery vendor. The COL applicant will specify the safety-related battery float voltage and equalizing voltage values, as described in COL Information Item 8.3-4-1-A.

The safety-related battery chargers are full-wave, silicon-controlled rectifiers. The housings are freestanding, National Electrical Manufacturers Association (NEMA) [Standards Publication 250-2003, “Enclosures for Electrical Equipment (1000 Volts Maximum)”] Type 1, and are ventilated. The chargers are suitable for float charging the batteries and operate from a 480-V, three-phase, 60-hertz (Hz) supply. The power for each divisional battery charger will be supplied by that division’s dedicated IPC. Each battery charger will be capable of recharging its battery from the design minimum charge to a fully charged condition within 24 hours, while supplying the full load associated with the individual battery. The battery chargers are the constant voltage type, adjustable between 240 V and 290 V, with the ability to operate as battery eliminators.

The battery eliminator feature will be incorporated as a precautionary measure to protect against the effects of inadvertent disconnection of the battery. The battery chargers are designed to function properly and remain stable if the battery is disconnected. Variation of the charger output voltage is less than ±1 percent, with or without the battery connected. The maximum output ripple for the charger is 30 millivolt root mean square with the battery, and less than 2 percent root mean square without the battery.

The battery chargers’ output will have a current-limiting design. The chargers are designed to prevent their ac source from becoming a load on the batteries because of power feedback from a loss of ac power. The battery chargers’ output voltage will be protected against overvoltage by a high-voltage shutdown circuit. When high voltage occurs, the unit disconnects the auxiliary voltage transformer, which results in charger shutdown. An alarm in the MCR indicates the loss of charger input voltage and charger shutdown.

Ventilation

A safety-related ventilation system will not be required for the batteries to perform their safety-related functions. However, battery rooms are ventilated by a system designed to remove the hydrogen gas produced during the charging of batteries. The system will be designed to preclude the possibility of hydrogen accumulation as described in DCD Tier 2, Revision 9, Section 9.4.6.

Monitoring and Alarms

Important system components are either self-alarming on failure or capable of clearing faults, or they are being tested during service to detect faults. All abnormal conditions of important system parameters, such as system grounds, charger failure, and low bus voltage, are alarmed in the MCR, locally, or both.

Inspection, Maintenance, and Testing of Direct Current System

An initial composite test of the onsite dc power system will be a prerequisite to initial fuel loading. This test verifies that each battery capacity will be sufficient to satisfy a design-basis load demand profile under the conditions of a LOCA and a LOPP. Conducted in accordance with IEEE Std 450, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” these tests ensure that the battery has the capacity to meet safety-related load demands.

The applicant stated that the ESBWR technical specifications (TS) describe the inservice tests, inspections, and maintenance of the dc power systems, including the batteries, chargers, and auxiliaries.

8.3.2.2.2 Nonsafety-Related Direct Current Systems

The nonsafety-related dc systems consist of five divisions of 250 V and two divisions of 125 V. The dc systems are ungrounded for reliability. The 125-V batteries provide dc power for nonsafety-related loads. The 250-V battery bank provides dc power for the plant's nonsafety-related distributed control and instrumentation system (DCIS) and nonsafety-related dc motors. Each of the dc systems has a battery, battery charger, standby battery charger, main dc distribution bus, and ground detection panel, except the 250-V dc load groups A and B. The A and B load groups each have two normal battery chargers, one standby battery charger, two batteries, a ground detection panel, and two distribution buses. The main distribution buses feed the local dc distribution panels, UPS inverter, and dc MCC. The plant design and circuit layout of the nonsafety-related dc systems provide physical separation of the equipment, cabling, and instrumentation associated with the load groups of nonsafety-related equipment. Each 125-V and 250-V battery will be separately housed in a ventilated room apart from its charger, distribution panel, and ground detection panel. Equipment of each load group of the dc distribution system will be located in an area separated physically from the other load groups.

The 125-V nonsafety-related battery bank will be sized for 2-hour duty cycles at a discharge rate of 2 hours, based on a terminal voltage of 1.75-V per cell at 25 degrees C (77 degrees F). The dc system minimum battery terminal voltage at the end of the discharge period is 105 V. The maximum equalizing charge voltage for the 125-V batteries is specified by the battery vendor.

The 250-V nonsafety-related batteries are sized for 2-hour duty cycles at a discharge rate of 2 hours, based on a terminal voltage of 1.75 V per cell at 25 degrees C (77 degrees F). The dc system minimum battery terminal voltage at the end of the discharge period is 210 V. The maximum equalizing charge voltage for 250-V batteries is specified by the battery vendor.

The nonsafety-related batteries have sufficient stored capacity, without their chargers, to independently supply their loads continuously for at least 2 hours. The batteries are sized so that the sum of the required loads does not exceed the battery ampere-hour rating, or the warranted capacity at end-of-installed-life with 100-percent design demand. The battery banks are designed to permit replacement of individual battery cells.

The nonsafety-related battery chargers are full-wave, silicon-controlled rectifiers or an acceptable alternative design. The housings are freestanding, NEMA Type 1, and are ventilated. The chargers are suitable for float charging the batteries. The chargers operate from a 480-V, three-phase, and 60-Hz ac supply. A separate power center, backed up by the onsite standby DG, supplies each charger. Standby chargers are used to equalize battery charging. Standby chargers are supplied from a different power center than the normal battery charger.

The battery chargers are the constant-voltage type, with the 125-V dc system chargers having a voltage adjustable between 120 V and 145 V, and the 250-V dc system chargers having a voltage adjustable between 240 V and 290 V, with the capability of operating as battery eliminators. The battery eliminator feature will be incorporated as a precautionary measure to protect against the effects of inadvertent disconnection of the battery. The battery chargers are designed to function properly and remain stable when the battery bank is disconnected. Variation of the charger output voltage will be less than ±1 percent, with or without the battery bank connected.

The battery chargers are designed to be output-current limiting and to have protection against power feedback from the battery bank to the ac supply system. The battery charger will be equipped with overvoltage protection by a high-voltage shutdown circuit to protect equipment from damage caused by high voltage. An alarm in the MCR indicates a loss of voltage to the charger and charger shutdown. Battery rooms are ventilated by a system designed to remove the hydrogen gas produced during the charging of batteries. The design of the system precludes the possibility of hydrogen accumulation.

8.3.2.3 *Staff Evaluation*

The safety-related batteries are specified in DCD Tier 2, Revision 3, to have sufficient capacity, without their chargers, to independently supply the safety-related loads continuously for at least 72 hours. Batteries must be sized for the dc load, in accordance with IEEE Std 485, "IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Nuclear Power Generating Stations," with an expected 20-year service life. In RAI 8.3-49 and RAI 8.3-52 and its supplements, the staff requested the loading profile to evaluate whether the safety-related 250-V batteries are of a size sufficient to meet the design requirements of their connected loads, without the charger support, for the corresponding period of 72 hours. In response to RAI 8.3-49 and RAI 8.3-52 S01, the applicant indicated that the final loading profile will not be determined until the DCIS loads are established during procurement. Instead, the inspection, maintenance, and testing program states that battery capacity tests will be conducted in accordance with IEEE Std 1188, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Valve-Regulated Lead-Acid (VRLA) Batteries for Stationary Applications." These tests ensure that the battery has the capacity to meet safety-related load demands. The final load profile will include the analyses and will be tested in accordance with ITAAC, Table 2.13.3-1, Items 3a and 3b, "Acceptance Criteria."

3a. Analyses reports of the as-built batteries exist and conclude that two sets of safety-related batteries in each division have the capacity, as determined by the vendor performance specification, to supply its rated constant current, for a minimum of 72 hours without recharging.

3b. Test reports conclude that the capacity of each as-built safety-related battery equals or exceeds the analyzed battery design duty cycle capacity.

The applicant also indicated that the safety-related batteries are sized to meet the design requirements of their connected load, without the charger support, for the corresponding period of 72 hours. A preliminary battery size has been selected to meet the estimated maximum design load profile, with the ability to increase the battery size by 50 percent of the estimated battery size. The selected batteries are capable of being sized to meet the above-stated criteria without expansion of the current rooms designated for each division's batteries. The battery bank will be designed to replace any defective cells without an interruption of service. However, the staff found that the applicant did not provide the loading profile to demonstrate that the safety-related 250-V batteries are sized to meet the design requirement of their connected load for the corresponding time period of 72 hours without the charger's support.

In RAI 8.3-52 S03, the staff requested information regarding the batteries' capacity to meet the design requirement of their connected load for the corresponding time of 72 hours without the battery charger's support. RAI 8.3-52 was being tracked as open item in the SER with open items. In response to RAI 8.3-52 S03, the applicant added a new DCD Tier 2, Revision 5, Table 8.3-3 to provide the nominal load requirements for each safety-related division. The applicant also independently submitted a summary of the safety-related battery sizing calculation, including relevant parameters and factors used in the calculation. The load cycle is based on the conservative estimation of the safety-related UPS loads provided in DCD Tier 2, Revision 5, Table 8.3-3. The battery sizing calculation summary confirms that this load cycle incorporates applicable safety-related UPS loads and includes conservative adjustments for inverter efficiency and power factor. DCD Tier 2, Revision 5, Section 8.3.2.1.1, identifies the following additional considerations included in battery sizing: (1) the dc system's minimum battery terminal voltage at the end of the discharge period is 210 V dc (1.75 V per cell), (2) the batteries are sized for the dc load, in accordance with IEEE Std 485, with an expected 20-year service life, and (3) the batteries include a margin to compensate for uncertainty in determining the battery state of charge. The battery sizing calculation summary included these considerations, including the design margin, aging factor, and temperature correction factor, consistent with IEEE Std 485. The temperature correction factor is consistent with the minimum cell temperature of 60 degrees F (15.6 degrees C) specified in DCD Tier 2, Revision 6, Table 8.3-3, Note (3). The calculation also includes a factor for uncertainty in float current monitoring. The staff finds that the battery sizing calculation summary confirms that batteries, based upon the considerations for determining battery size identified in DCD Tier 2, Revision 6, Chapter 8 are sized adequately and sufficiently to supply uninterruptible power for 72 hours. Therefore, the staff concluded that the battery sizing is acceptable. However, the staff determined that additional battery capacity information was needed in the DCD.

In RAI 8.3-52 S04, the staff requested information regarding the battery capacity in ampere-hours, the specifications for the charger, rectifier, inverter, and regulating transformer, and the UPS protective scheme against faults. In its response, the applicant added a new Table 8.3-4 to DCD Tier 2, Revision 5, which includes nominal values for the battery capacity and the specifications for the charger, inverter, and regulating transformer. The applicant explained that the UPS rectifier specification is considered to be included as part of the overall UPS, as its sizing and specification are dictated by the UPS (inverter) specification found in the new Table 8.3-4. The applicant further stated that the safety-related UPS is protected against overvoltage, undervoltage, overfrequency, underfrequency, overcurrent, and fault current. The UPS protection features are integral to the unit and will be set based on the calculation performed as part of the ESBWR detailed design. Additionally, regulating transformers were deleted in response to RAI 8.2-14 S01, as discussed in Section 8.3.1.3 of this report. The staff found that the applicant provided the requested information and that the battery capacity, charger sizing, and inverter sizing are consistent with the dc load profile. The staff confirmed that Table 8.3-4 was included in Revision 5 of the DCD. Based on the applicant's response, RAIs 8.3-49 and 8.3-52 are resolved.

DCD Tier 2, Revision 5, Section 8.3.2.2.1, stated that the safety-related batteries meet the qualification requirements of IEEE Std 535, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations." Section 1.1, "Scope," of IEEE Std 535 states that the standard describes qualification methods for Class 1E VLA batteries and that consideration of other types of batteries are beyond the scope of the standard. Consequently, IEEE Std 535 does not apply to VRLA batteries. In RAI 8.3-63, the staff asked the applicant to explain how these batteries were going to be qualified, including both the methods used and the process flow. In its response, the applicant referred to the change of the ESBWR safety-related batteries to VLA, in response to RAI 8.3-62, which is discussed below. The method of qualifying VLA batteries is discussed in response to RAI 8.3-64. Based on the applicant's response, RAI 8.3-63 is resolved.

DCD Tier 2, Revision 5, Section 8.3.2.2.1, stated that the safety-related batteries meet the qualification requirements of IEEE Std 535, which was written under the assumption of an 8-hour duty cycle. IEEE Std 535 does not apply to duty cycles longer than 8 hours. In RAI 8.3-64, the staff asked the applicant to identify the methodology to be used to qualify these batteries for an extended duty cycle of 72 hours. Also, the staff asked the applicant to discuss the failure mode(s) for this type of battery for the 72-hour duty cycle. In addition, in RAI 8.3-65, the staff requested the applicant to justify a 20-year battery service life. In response to RAIs 8.3-64 and 8.3-65, the applicant stated that safety-related batteries are qualified to meet IEEE Std 535 by type test, with the exception that the duty cycle is 72 hours, and that supplemental discharge cycle testing is required to meet the harsh environment qualification process of IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Additionally, the equipment qualification process for batteries includes the evaluation of significant aging mechanisms that are related to failure mechanisms from radiation exposure, time-temperature aging, and cycle aging; age testing for significant aging mechanisms for a 20-year qualified life; seismic tests; and performance testing for the 72-hour duty cycle. The applicant submitted Licensing Topical Report, NEDE-33516P, "ESBWR Qualification Plan Requirements for a 72-Hour Duty Cycle Battery," July 27, 2009. The applicant provided a detailed testing plan in the topical report. The acceptability of the 72-hour duty cycle battery qualification plan is evaluated in the SER for NEDE-33516P. Based on the applicant's response and NEDE-33516P, RAIs 8.3-64 and 8.3-65 are resolved.

The following evaluation documents the compliance of the safety-related dc power supply systems with the regulatory criteria and their consistency with the guidance of the SRP. The ESBWR four-division design allows the systems to sustain a credible single active failure with one division already out of service and the remaining two divisions fully performing their safety function for 72 hours without chargers. The dc power supply systems comply with GDC 2, 4, 17, 18, 33, 34, 35, 38, 41, 44, and 50, and with 10 CFR 50.63 and 10 CFR 50.55a(h), based on conformance with the following RGs:

- RG 1.6

  In RAI 8.3-61, the staff asked the applicant to revise the section related to RG 1.6 and discuss how the design meets the RG 1.6 regulatory positions. In its response, the applicant stated that the applicability statement found in DCD Tier 2, Revision 5, Section 8.3.2.2.2, regarding RG 1.6 had an editorial error that stated that RG 1.6 for "Standby (Onsite) Power Sources" is not applicable to the passive ESBWR design. The applicant revised the applicability text at Section 8.3.2.2.2 for RG 1.6. Based on the applicant's response, RAI 8.3-61 is resolved. The staff confirmed that Revision 6 of the DCD includes the change described above. The dc power systems are designed with required independence. Based on the above, the staff finds that the design is consistent with the guidance of RG 1.6.

- RG 1.32

  The design provides for safety-related dc power supply systems to support passive core cooling and containment integrity safety functions. The design is consistent with the guidance of RG 1.32.

- RG 1.53

  The safety-related dc system is designed so that no single active failure in any division of the 250-V dc system results in conditions that prevent the safe shutdown of the plant while a separate division is out of service for maintenance. Therefore, the design is consistent with the guidance of RG 1.53.

- RG 1.63

  Redundant overcurrent interrupting devices are provided for electrical circuits routed through containment penetrations if the maximum available fault current is greater than the continuous rating of the penetration. Therefore, the design is consistent with the guidance of RG 1.63.

- RG 1.75

  Safe shutdown relies upon dc-derived power and meets the design requirements for physical independence. Therefore, the design is consistent with the guidance of RG 1.75.

- RG 1.81

  The ESBWR plant will be designed as a single-unit plant, and, thus, GDC 5 and RG 1.81 are not applicable.

- RG 1.128 and RG 1.129

  In RAI 8.3-62, the staff asked the applicant to explain how these two RGs are applicable to VRLA batteries. In its response, the applicant revised the DCD to reflect a change for the safety-related batteries to VLA. Therefore, RG 1.128 and RG 1.129 apply to "large lead storage batteries," and their endorsed IEEE standards are applicable to the ESBWR design. The applicant also revised DCD Tier 2, Revision 5, to state that the maximum equalizing charge voltage for safety-related batteries will be site specific and that the COL applicant will specify the safety-related battery float voltage and equalize voltage. The applicant added COL Information Item 8.3.4-1A. The applicant also added two notes (Notes 3 and 4) to Table 8.3-3 to state that 60 degrees F (15.6 degrees C) will be the minimum operable temperature used in the sizing calculation for the safety-related batteries and that the battery sizing calculation used the methodology of IEEE 485, which includes an overall margin that will be conservative and bounding. Based on the applicant's response, RAI 8.3-62 is resolved. The staff confirmed that Revision 6 to the DCD includes the change described above. The staff finds that the battery sizing calculation will be conservative and hence, acceptable. The staff finds that the design is consistent with the guidance of RG 1.128 and RG 1.129.

- RG 1.118

  The ESBWR TSs include in-service tests and inspections and the resulting maintenance of the dc power systems, including the batteries, chargers, and auxiliaries. Therefore, the design is consistent with the guidance of RG 1.118.

- RG 1.153

  Safe shutdown relies upon dc-derived power and meets the design requirements for redundancy. The redundant safety-related loads are distributed between redundant distribution systems and power systems are supplied from the related redundant distribution systems. Therefore, the design is consistent with the guidance of RG 1.153.

- RG 1.155

  The ESBWR uses battery power to achieve and maintain safe shutdown. The safety-related batteries have sufficient stored capacity, without their chargers, to independently supply the safety-related loads continuously for 72 hours. Thus, the ESBWR meets the intent of RG 1.155. (See the SBO evaluation in Sections 8.4.2.1 and 15.5.5 of this report.)

- RG 1.160

  As described in DCD Tier 2, Revision 9, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

- RG 1.182

  As described in DCD Tier 2, Revision 9, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

The ESBWR plant is designed as a single-unit plant, and, thus GDC 5 is not applicable. As described in DCD Tier 2, Revision 9, Section 17.4.1, for 10 CFR 50.65(a)(4), the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

8.3.2.4 *Combined License Unit-Specific Information*

The applicant stated that the COL applicant will address safety-related battery float and equalizing voltage values via COL Information Item 8.3.4-1-A. This COL item identifies an item related to the dc power system that the COL applicant will address. The staff agrees that this item is site-specific and therefore appropriately addressed by the COL applicant. In addition, this COL item for the dc power system, provided it is adequately addressed, provides reasonable assurance that the dc power system meets the requirements of GDC 17. Therefore, the staff finds this COL information item acceptable.

8.3.2.5 *Conclusion*

Based on its review, the staff concludes that the applicant has provided sufficient information to demonstrate that the onsite dc power supply systems meet applicable regulatory requirements and are capable of providing the power supply to onsite loads needed to support the plant's safe operation. The design of the onsite dc power supply systems is acceptable and meets the requirements of GDC 2, 4, 17, 18, 33, 34, 35, 38, 41, 44, and 50, and of 10 CFR 50.63 and 10 CFR 50.55a(h).

8.4 <u>Safety Analysis Issues</u>

8.4.1 Generic Issues and Operational Experience

8.4.1.1 *Technical Evaluation*

The staff evaluated the generic issues (GI) (Task Action Plan items and new GIs identified in NUREG–0933, "Resolution of Generic Safety Issues," issued August 2008) and operational experience (GLs and BLs) described in the sections below.

8.4.1.1.1 Task Action Plan Items

The staff evaluated the following four Task Action Plan items:

1. A-25, "Nonsafety Loads on Class 1E Power Sources"—If nonsafety-related loads are allowed to be connected to the Class 1E power system, it is possible that they may cause degradation by introducing loss of redundancy or another failure mechanism. The 120-V ac emergency lighting in the MCR and in the remote shutdown area is non-Class 1E and is fed from a Class 1E UPS, through series isolation devices that are coordinated with upstream 120-V ac distribution panel circuit breakers. The ESBWR design meets RG 1.75 as discussed in Section 8.3.1.3 of this report. RG 1.75 allows the connection of nonsafety loads to Class 1E (emergency) power sources, if it can be shown that the connection of nonsafety loads will not result in the degradation of the Class 1E system. In the ESBWR design, either of these protective devices is able to interrupt any fault current before initiation of a trip of any upstream protective device. No failure of non-Class 1E equipment or system will degrade the Class 1E system below an acceptable level. Therefore, the issue is resolved.

2. A-30, "Adequacy of Safety-Related dc Power Supplies"—New GI 128 addresses the reliability of onsite electrical systems and encompasses GI A-30. See the staff discussion below on GI 128.

3. A-35, "Adequacy of Offsite Power Systems"—As applied to the ESBWR, Task Action Plan Item A-35 is associated with minimizing the likelihood of simultaneous failure of both the offsite power supply circuits. The offsite power system of the ESBWR plant is based on the following design bases: Two independent and physically separate offsite circuits supply reliable power to the plant. Electric power from the utility grid to the offsite power system is provided by transmission lines designed and located to minimize the likelihood of failure while ensuring grid reliability. The transmission systems supply the two offsite power circuits through their respective transformers in the switchyard. Any single active failure can affect only one power supply and cannot propagate to the alternate power supply. In addition, the ESBWR design does not require any offsite ac power to achieve and maintain safe shutdown for 72 hours. Therefore, this issue is resolved.

4. A-44, "Station Blackout"—GI A-44 was resolved with the publication of 10 CFR 50.63, which requires that LWRs be able to withstand, for a specified duration, and recover from, an SBO. It addresses the likelihood of the loss of all ac power at the site and the potential for severe core damage after an SBO. The ESBWR is designed to shut down safely without reliance on offsite or DG-derived ac power for 72 hours, which exceeds SBO requirements. Section 8.4.2.1 of this report discusses in detail how the design successfully addresses this issue. Therefore, this issue is resolved for the ESBWR standard plant design.

8.4.1.1.2 New Generic Issues

The staff addressed the following two new GIs:

1. GI 128, "Electrical Power Reliability"—GI 128 addresses the reliability of onsite electrical systems and encompasses GI 48, "LCO for Class 1E Vital Instrumentation Buses in Operating Reactors"; GI 49, "Interlocks and LCO for Class 1E Tie Breakers"; and GI A-30, "Adequacy of Safety-Related dc Power Supplies." The staff has reviewed the applicant's

submittal and concludes that the ESBWR design addresses GIs 48, 49, and A-30 for the following reasons:

- GI 48—The applicant provided the limiting condition for operation (LCO) in the event of a loss of one or more Class 1E 120-V ac vital instrument buses and associated inverters. The staff finds this LCO acceptable.
- GI 49—The ESBWR design does not include Class 1E tie breakers.
- GI A-30—The staff has evaluated the Class 1E dc distribution system design for the aspects addressed by GI A-30 in Section 8.3.2 of this report and concludes that it is acceptable.

Therefore, GI 128 is resolved for the ESBWR design.

2. GI 107, “Main Transformer Failures”—As applied to the ESBWR, GI 107 is associated with minimizing the effects of transformer oil leaks and resulting fires. The ESBWR has design features to address this issue. Three-hour fire rated concrete barriers are used between the RATs, the UATs and the main transformers and spare main transformer. Pits are provided for the containment and collection of transformer oil should it leak. Also, the main transformers are included in the ESBWR fire hazard analysis in the DCD Tier 2, Revision 9, Section 9A.4.7, which describes the fire protection systems for the main transformer. The staff finds that these systems are consistent with the guidelines of GI 107. Therefore, this issue is resolved.

8.4.1.1.3 Generic Letters

The staff addressed the following five GLs:

1. GL 84-015, “Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,” dated July 2, 1984—This item is not applicable to the ESBWR design, since it does not require a safety-related emergency diesel generator (EDG).

2. GL 88-015, “Electric Power System—Inadequate Control over Design Processes,” dated September 12, 1988—This GL informs the licensees of the various problems with electrical systems occurring with increasing frequency at nuclear power plants. These problems include onsite distribution system voltages lower than required for proper operation of safety equipment, EDG loading exceeding design, inadequate EDG response to actual loading, overloading Class 1E buses, inadequate breaker coordination, and inadequate fault current interruption capability. Problems associated with EDGs are not applicable to the ESBWR design, since it does not require an EDG. DCD Tier 1, Revision 9, Table 2.13.1-2, discusses the voltage adequacy, breaker coordination, and fault current interrupting capability. DCD Tier 2, Revision 9, Section 8.3.2 discusses degraded voltage and overvoltage. Based on the above, this GL is resolved for the ESBWR design.

3. GL 91-006, “Resolution of Generic Issue (GI) A-30—‘Adequacy of Safety-Related dc Power Supplies,’ Pursuant to 10 CFR 50.54(f),” issued in 1991—GI A-30 is integrated into GI 128. See the staff discussion above on GI 128.

4. GL 91-011, “Resolution of GI 48, ‘LCOs for Class 1E Vital Instrument buses,’ and GI 49, ‘Interlocks and LCOs for Class 1E Tie Breakers,’ Pursuant to 10 CFR 50.54 (f),” dated July 8, 1991—GI 48 and GI 49 are integrated into GI 128. See the staff discussion above on GI 128.

5. GL 94-001, “Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,” dated May 31, 1994—This item is not applicable to the ESBWR design, since the design does not require an EDG.

8.4.1.1.4 Bulletin

The issue addressed in BL 82-04, “Deficiencies in Primary Containment Electrical Penetration Assemblies,” dated December 3, 1982, applies to a specific equipment supplier that is no longer selling primary containment electrical penetration assemblies. The ESBWR will use primary containment electrical penetration assemblies that are qualified to IEEE Std 317 requirements, in accordance with RG 1.63. DCD Tier 2, Revision 9, Section 3.11, discusses the qualification of primary electrical penetrations. Therefore, the issue is resolved.

8.4.1.2 *Conclusion*

Based on the above discussion, the staff concludes that the GIs and operational experience issues are resolved for the ESBWR design.

8.4.2 Advanced Light-Water Reactor Certification Issues

The following sections discuss the policy, technical, and licensing issues pertaining to passive designs that relate to the electrical portion of the ESBWR design.

8.4.2.1 *Station Blackout*

The term SBO refers to the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. An SBO, therefore, involves the loss of the offsite electric power system (“preferred power system”), concurrent with a turbine trip and the unavailability of the emergency alternating current (EAC) power system. An SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources specifically provided for SBO mitigation. Because many safety systems necessary for reactor core decay heat removal depend on ac power, an SBO could result in a severe core damage accident. The risk of SBO involves the likelihood and duration of the loss of all ac power and the potential for severe core damage after a loss of all ac power. DCD Tier 2, Revision 9, Section 15.5.5, discusses SBO.

8.4.2.1.1 Regulatory Criteria

The acceptance criteria for evaluating whether a plant is capable of withstanding and recovering from an SBO are based on meeting the relevant requirements of the following regulations:

- GDC 17, as it relates to (1) the capacity and capability of onsite and offsite power systems to permit the functioning of SSCs important to safety in the event of anticipated operational occurrences and postulated accidents and (2) provisions to minimize the probability of losing electric power from the transmission network (grid) as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies

- GDC 18, as it relates to periodic testing and inspection of offsite and onsite power systems important to safety

- 10 CFR 50.63, as it relates to the capability to withstand and recover from an SBO

- 10 CFR 50.65(a)(4), as it relates to the assessment and management of the increase in risk that may result from proposed maintenance activities before performing the maintenance activities, including, but not limited to, surveillances, post-maintenance testing, and corrective and preventive maintenance. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17.

SRP Section 8.4 specifies that an application meets the above requirements if the application satisfies the following guidance:

- The guidelines of RG 1.155, as they relate to compliance with the requirements of 10 CFR 50.63

- The guidelines and criteria of SECY-90-016 and SECY-94-084, as they relate to the use of alternate ac power sources and RTNSS at plants provided with passive safety systems

- The guidelines of RG 1.9 and RG 1.155, as they relate to the reliability program implemented to ensure that the target reliability goals for onsite EAC power sources (typically, DG units) are adequately maintained

- The guidelines of RG 1.160, as they relate to the effectiveness of maintenance activities for onsite EAC power sources, including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO)

- The guidelines of RG 1.182, as they relate to conformance with the requirements of 10 CFR 50.65(a) (4) for assessing and managing risk when performing maintenance

8.4.2.1.2 Summary of Technical Information

DCD Tier 2, Revision 9, Section 15.5.5, documents the ESBWR SBO analysis. As a passive plant, the ESBWR does not rely on ac power to achieve hot or stable shutdown. With regard to electrical power topics, the SBO evaluation assumes that the loss of all ac power occurs at time zero. The evaluation also shows that Q-DCIS provides control power, closure, and position indication for containment isolation valves. DCD Tier 2, Revision 9, Section 8.3.2.1.1, describes the power supply.

8.4.2.1.3 Staff Evaluation

DCD Tier 2, Revision 9, Section 15.5.5, documents the ESBWR SBO analysis. This section describes the criteria, assumptions, and analyses used to show that the ESBWR can cope with SBO conditions without ac power for 72 hours. Section 15.5.5 of this report provides the staff evaluation of the ESBWR SBO analysis. This section contains the Chapter 8 staff evaluation applicable to passive plants.

The requirements of 10 CFR 50.63 state that “each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout.” The ESBWR design will not have EAC power sources, and it is not required to evaluate SBO coping duration, as long as the design will be capable of performing safety-related functions for 72 hours without ac power. An alternate ac power source is not necessary for passive plant designs that (1) do not need ac power to perform safety-related functions for 72 hours following the onset of an SBO and (2) meet the guidelines in Section C.IV.9 of RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007, regarding RTNSS (refer to Chapter 22 of this report for RTNSS). The ESBWR design minimizes the potential risk contribution of an SBO by not relying on the ac power supply to perform safety-related functions.

The following paragraphs analyze compliance with the regulatory criteria:

- GDC 17

  With regard to GDC 17, the ESBWR plant design does not require offsite or diesel-generated ac power for 72 hours after an SBO event. However, the offsite ac power meets GDC 17 (see the staff evaluation in Section 8.2 of this report). Safety-related dc power supports passive core cooling and containment safety-related functions for 72 hours. Safety-related dc power meets GDC 17 (see the staff evaluation in Section 8.3.2 of this report).

- GDC 18

  Safety-related dc power supports passive core cooling and containment safety-related functions. No offsite or diesel-generated ac power is required for 72 hours after an abnormal event. Safety-related dc power meets GDC 18 (see the staff evaluation in Section 8.3.2 of this report).

- 10 CFR 50.63 and RG 1.155

  Section 15.5.5 of this report evaluated the ESBWR coping capability with an ac-independent approach and finds it to be acceptable. Therefore, the ESBWR design also meets the requirements of 10 CFR 50.63.

  In RAI 8.1-21, the staff asked the applicant to address procedures and training to cope with an SBO. In its response, the applicant stated that, after 72 hours, required loads will be powered from the ancillary DGs, or the standby DGs or offsite power, if available. This is consistent with the ancillary diesels being designated as RTNSS, as described in DCD Tier 2, Revision 9, Appendix 19 A. Emergency procedures will cover the starting and connection of the diesels and restoration of ac power, as required by Section 3.4 of RG 1.155. The development of procedures is described in DCD Tier 2, Revision 9, Section 13.5.2, and is covered by the COL information items found in DCD Tier 2, Revision 9, Section 13.5.3. Training is described in DCD Tier 2, Revision 9, Section 13.2, and COL information items for training are covered in DCD Tier 2, Revision 9, Section 13.2.5. However, these sections do not specifically address SBO events. Instead, they commit to procedures and training for all of the plant's normal, abnormal, and emergency events. This would include SBO. To address this, the applicant added a note in DCD Tier 2, Revision 6, Table 8.1-1, to state that procedures and training for SBO response guidelines, ac power restoration, and severe weather guidelines are developed per Sections 13.2 and 13.5. The staff finds that the applicant adequately addressed the issue and, therefore, RAI 8.1-21 is resolved. The staff confirmed that Revision 6 of the DCD includes the change described above. The staff finds that the description of SBO procedures and training are consistent with the guidelines of RG 1.155.

- SECY-90-016

  This paper contains the Commission's approval for the evolutionary ALWRs to have an alternate ac power source of diverse design capable of powering at least one complete set of normal shutdown loads to cope with SBO. This topic is not applicable to the ESBWR design, since no ac power is required to achieve safe shutdown.

- SECY-94-084 and SECY-95-132

  Section 8.4.2.3 and Section 22 of this report discuss this topic.

- 10 CFR 50.65(a) (4), RG 1.160, and RG 1.182

  As described in DCD Tier 2, Revision 9, Section 17.4.1, the maintenance rule program is part of the operational reliability assurance program covered by COL Information Item 17.4-2-A, which is evaluated in Section 17.4 of this report.

- RG 1.9 and RG 1.155

  RG 1.9 and RG 1.155 regarding a reliability program for emergency onsite ac power source are not applicable.

8.4.2.1.4 Conclusion

The ESBWR reactor core and associated coolant, control, and protection systems, including station batteries and other necessary support systems, provide sufficient capacity and capability to ensure that the core will be cooled and will have appropriate containment integrity for 72 hours in the event of an SBO. The staff concludes that the safety-related passive systems are capable of withstanding a loss of all ac power for 72 hours. The staff further concludes that the ESBWR design will be in compliance with the provisions of GDC 17 and 18 and 10 CFR 50.63, as they relate to the capability to achieve and maintain hot or stable shutdown in the event of an SBO.

8.4.2.2 *Electrical Distribution*

8.4.2.2.1 Technical Evaluation

The Commission approved the following recommendations in SECY-91-078 for plant designs:

1. An alternate offsite power source will be available for nonsafety-related loads, unless the design margins for loss of nonsafety-related loads are no more severe than turbine-trip-only events in current plants.

2. At least one offsite circuit to each redundant safety division will be supplied directly from offsite power sources, with no intervening nonsafety-related buses.

The medium-voltage PG and PIP buses feed nonsafety-related loads in the ESBWR design. These buses are usually powered from the normal preferred power source through the UATs. These buses are also capable of being powered from the alternate preferred power source RATs through an automatic bus transfer, in the event that the normal preferred power source is unavailable. Thus the ESBWR design meets recommendation 1. The ESBWR design does not have to meet recommendation 2, because the design does not rely on active systems for safe shutdown.

8.4.2.2.2 Conclusion

The ESBWR design meets recommendation 1. The ESBWR design does not have to meet recommendation 2, because the design does not rely on active systems for safe shutdown.

8.4.2.3 *Regulatory Treatment of Nonsafety Systems*

8.4.2.3.1 Technical Evaluation

The staff considered whether the applicant identified RTNSS functions and availability controls for electrical systems, consistent with the Commission's policy in SECY-95-132.

DCD Tier 2, Revision 9, Table 19.A-2, shows that the onsite standby DGs and the 6.9-kV PIP buses have RTNSS functions. The standby DGs supply ac power to the PIP nonsafety-related buses. The PIP buses feed nonsafety-related loads required for a unit's normal operation and shutdown. In addition, the PIP nonsafety-related buses supply ac power to the safety-related IPCs. The standby DGs are required to provide power for recharging batteries to support post-accident monitoring and the fuel and auxiliary pools cooling system (FAPCS). In addition, the standby DGs provide power to the reactor water cleanup/shutdown cooling (RWCU/SDC) system operating in the shutdown cooling mode in the event of a loss of preferred power. The staff finds the onsite standby DGs and the 6.9-kV PIP buses RTNSS functions acceptable.

Additionally, DCD Tier 2, Revision 9, Table 19.A-2, shows that the ancillary DGs have RTNSS functions. The two nonsafety-related, seismic Category II ancillary DGs provide 480-V ac power for post-accident support loads when the normal and alternate preferred 6.9-kV power supplies and the standby DGs are not available. The staff finds the ancillary DGs RTNSS functions acceptable.

The availability controls require that one standby DG, its auxiliary systems (fuel storage tank and transfer system), and PIP buses be available during Modes 1, 2, 3, and 4 to support FAPCS and the ability to recharge batteries to support post-accident monitoring. Two standby DGs, their auxiliary systems (fuel storage tank and transfer system), and PIP buses are required to be operable during Modes 5 and 6 when core heat will be removed by the RWCU/SDC system. Planned maintenance should not be performed on standby DGs during operation in Modes 5 and 6. The availability controls require that two ancillary DGs with fuel tanks, fuel oil transfer pumps, and ancillary buses be available during all modes of plant operation. The staff finds that the applicant has provided acceptable availability controls for the electrical systems that have RTNSS functions and has included these controls in the DCD.

8.4.2.3.2 Conclusion

The staff reviewed the RTNSS designation and availability controls of electrical systems and finds them acceptable. Chapter 22 of this report provides additional discussion on RTNSS.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This final safety evaluation report documents the technical review of General Electric-Hitachi's (GEH's) Economic Simplified Boiling-Water Reactor (ESBWR) design certification. GEH submitted its application for the ESBWR design on August 24, 2005, in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52. The NRC formally docketed the application for design certification (Docket No. 52-010) on December 1, 2005. The ESBWR design is a boiling-water reactor (BWR) rated up to 4,500 megawatts thermal (MWt) and has a rated gross electrical power output of 1,594 megawatts electric (MWe). The ESBWR is a direct-cycle, natural circulation BWR that relies on passive systems to perform safety functions credited in the design basis for 72 hours following an initiating event. After 72 hours, non-safety systems, either passive or active, replenish the passive systems in order to keep them operating or perform post-accident recovery functions directly. The ESBWR design also uses nonsafety-related active systems to provide defense-in-depth capabilities for key safety functions provided by passive systems. The ESBWR standard design includes a reactor building that surrounds the containment, as well as buildings dedicated exclusively or primarily to housing related systems and equipment. On the basis of its evaluation and independent analyses, as set forth in this report, the NRC staff concludes that GEH's application for design certification meets the requirements of 10 CFR Part 52, Subpart B, that are applicable and technically relevant to the ESBWR design. Appendix F includes a copy of the report by the Advisory Committee on Reactor Safeguards, as required by 10 CFR 52.53.

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