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NUREG-1503, Supplement 2

# Final Safety Evaluation Report

## Related to the Certification of the Advanced Boiling Water Reactor Design

## Supplement 2

Office of Nuclear Reactor Regulation

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# ABSTRACT

Appendix A, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," constitutes the standard design certification (DC) for the U.S. Advanced Boiling-Water Reactor (ABWR) design. To document the U.S. Nuclear Regulatory Commission (NRC) staff's review supporting initial certification of the ABWR, the staff issued a final safety evaluation report (FSER) in NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," in July 1994 and NUREG-1503, Supplement 1, in May 1997.

In a letter dated December 7, 2010 (Agencywide Documents Access and Management System Accession No. ML110040176), GE-Hitachi Nuclear Energy (GEH or the applicant) submitted a Design Certification (DC) renewal application for the ABWR pursuant to the requirements of Subpart B, "Standard Design Certifications," 10 CFR Part 52.

This supplemental FSER[1] (Supplement 2 to NUREG-1503) documents the NRC or the Commission staff's technical review.

GEH's renewal application includes the ABWR Design Control Document (DCD), Revision 7. The staff completed the review of the ABWR renewal DCD, Revision 7, and issued a supplemental FSER. The staff is planning to perform a direct final rule to renew the certification for the ABWR standard design.

The ABWR design is a single-cycle, forced-circulation, boiling-water reactor (BWR), with a rated power of 3926 megawatts thermal, originally designed by GE. The original design incorporated updated safety enhancements from previous GE BWRs including a reinforced concrete reactor containment vessel with built-in liner, reactor coolant recirculation system driven by internal pumps, advanced electric/hydraulic control rod drives using a screw mechanism, and integrated digital control system and instrumentation.

The renewed ABWR DC incorporates modifications related to aircraft impact analyses in accordance with 10 CFR § 52.59(a), which requires that the renewed DC complies with the applicable requirements of 10 CFR § 50.150, "Aircraft impact assessment." In addition, GEH incorporated updated emergency core cooling suction strainers, a size correction to the containment overpressure protection system, Fukushima-related safety enhancements, including an additional ac-independent water makeup system with external connections for water addition, ac power, and safety-related wide range spent fuel pool instrumentation.

On the basis of the staff's review of the application, as documented in this FSER, the staff recommends that the Commission approve the DC renewal of the ABWR design.

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[1] This FSER documents the NRC staff's position on all safety issues associated with the ABWR DC Renewal application. The Advisory Committee on Reactor Safeguards (ACRS) independently reviewed those aspects of the application that concern safety, as well as the advanced safety evaluation report without open items (an earlier version of this document) and provided the results of its review to the Commission in a report dated October 31, 2019. This report is included as Appendix E to this SER.

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# ABBREVIATIONS AND ACRONYMS

| | |
|---|---|
| ac | alternating current |
| ABWR | advanced boiling-water reactor |
| ACI | American Concrete Institute |
| ACIWA | ac-independent water addition |
| ACS | atmospheric control system |
| ACRS | Advisory Committee on Reactor Safeguards |
| ADAMS | Agencywide Documents Access and Management System |
| AEA | Atomic Energy Act |
| AIA | Aircraft Impact Assessment |
| AISI | American Iron and Steel Institute |
| ALARA | as low as reasonably achievable |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| AOO | anticipated operational occurrence |
| AOV | air-operated valve |
| AR | annual reporting letters |
| ASME | American Society of Mechanical Engineers |
| ASCE | American Society of Civil Engineers |
| ASCE/SEI | American Society of Civil Engineers/Structural Engineering Institute |
| ASTM | American Society of Testing and Materials |
| ATWS | anticipated transient without scram |
| BDBE | beyond-design-basis event |
| BE | best estimate |
| BL | Bulletin |
| BPV | Boiler and Pressure Vessel |
| BPVC | Boiler and Pressure Vessel Code |
| BTP | branch technical position |
| Btu | British thermal unit |
| BWR | boiling-water reactor |
| BWROG | Boiling Water Reactors Owners Group |
| C | celsius |
| cc | cubic-centimeters |
| C/B | control building |
| CDF | core damage frequency |
| CDI | conceptual design information |
| CENA | central and eastern North America |
| CFR | *Code of Federal Regulations* |
| cm | centimeter(s) |
| COL | combined license |
| COPS | containment overpressure protection system |
| CP | construction permit |
| CR | control room |
| CRD | control rod drive |
| CRHA | control room habitability area |
| CST | condensate storage tank |

| | |
|---|---|
| CW | circulating water |
| DAC | design acceptance criteria |
| DAW | dry active waste |
| DB | dry bulb |
| DBA | design-basis accident |
| DBE | design-basis event |
| DBT | design basis tornado |
| DC | design certification |
| dc | direct current |
| DCA | design certification application |
| DCD | design control document |
| DCR | design certification rule |
| DG | diesel generator |
| DLF | dynamic load factors |
| DNB | departure from nucleate boiling |
| DW-WW | drywell-wetwell |
| | |
| EAB | exclusion area boundary |
| EAC | emergency alternating current |
| EAL | emergency action level |
| ECCS | emergency core cooling system |
| EDG | emergency diesel generator |
| ELAP | extended loss of alternating current power |
| EMDG | extensive damage mitigation guideline |
| EMI | electromagnetic interference |
| EMS | essential multiplexing system |
| ENS | emergency notification system |
| EOC | emergency operations center |
| EOF | emergency operations facility |
| EOL | end of life |
| EOP | emergency operating procedure |
| EP | emergency planning |
| EP | emergency preparedness |
| EPA | Environmental Protection Agency |
| EPG | emergency procedure guideline |
| EPIP | emergency plan implementing procedure |
| EPRI | Electric Power Research Institute |
| EPZ | emergency planning zone |
| EQ | environmental qualification |
| ER | environmental report |
| ERDS | emergency response data system |
| ERF | emergency response facility |
| ERO | emergency response organization |
| ESBWR | economic simplified boiling-water reactor |
| ESF | engineered safety feature |
| ESP | early site permit |
| ETS | emergency telecommunications system |
| | |
| F | Fahrenheit |
| FB | fuel building |
| FEIS | Final Environmental Impact Statement |

| | |
|---|---|
| FEMA | Federal Emergency Management Agency |
| FERC | Federal Energy Regulatory Commission |
| FHA | fire hazards analysis |
| FLEX | diverse and flexible coping strategy |
| FP | fire protection |
| FPC | fuel pool cooling and cleanup |
| FPS | fire protection system |
| FR | Federal Register |
| FS | factor of safety |
| FSAR | final safety analysis report |
| FSER | final safety evaluation report |
| Ft | feet/foot |
| FWLB | feedwater line break |
| | |
| GDC | general design criterion/criteria |
| GE | General Electric |
| GEH | General Electric – Hitachi (Nuclear Energy) |
| GENE | General Electric Nuclear Energy (original ABWR applicant) |
| GI | generic issue |
| GL | generic letter |
| GMRS | ground motion response spectrum |
| gpm | gallons per minute |
| GSI | generic safety issue |
| GWMS | gaseous waste management system |
| | |
| hr | hour |
| HCLPF | high confidence of low probability of failure |
| HCU | hydraulic control unit |
| HEPA | high-efficiency particulate air |
| HF | high frequency |
| HFE | human factors engineering |
| HVAC | heating, ventilation, and air conditioning |
| HWL | high water level |
| Hz | Hertz |
| HX | heat exchanger |
| | |
| I&C | instrumentation and control |
| IBC | International Building Code |
| IDLH | immediate danger to life and health |
| IE | inspection and enforcement |
| IEEE | Institute of Electrical and Electronic Engineers |
| IN | information notice |
| in. | inch(es) |
| IOZ | inorganic zinc |
| ISFSI | independent spent fuel storage installation |
| ISG | interim staff guidance |
| ISI | inservice inspection |
| ISRS | in-structure response spectra |
| IST | inservice testing |
| ITAAC | Inspections, tests, analyses, and acceptance criteria |
| ITP | initial test program |

| | |
|---|---|
| JLD | Japan lesson-learned project directorate |
| | |
| km | kilometer(s) |
| kPa | kilopascals |
| kV | kilovolt |
| | |
| LAN | local area network |
| LB | lower-bound |
| lb/ft2 | pounds per square-foot |
| LCO | limiting condition for operation |
| LF | low frequency |
| LLRW | low-level radioactive waste |
| LOCA | loss-of-coolant accident |
| LOLA | loss of large area |
| LOOP | loss-of-offsite power |
| LOPP | loss of preferred power |
| LPFL | low pressure core flooder |
| LPZ | low population zone |
| LR | lower-range |
| LTR | licensing topical report |
| LWMS | liquid waste management system |
| LWR | light-water reactor |
| | |
| m | meter(s) |
| M&TE | measuring and test equipment |
| MBDBE | mitigation of beyond design basis events |
| MBtu | one million BTU |
| MC&A | material control and accounting |
| MCR | main control room |
| MCWB | mean coincident wet-bulb |
| MEI | maximally exposed individual |
| mi | mile(s) |
| MJ | megajoules |
| MOV | motor-operated valve |
| MPa | megapascals |
| MPaG | megapascals gauge |
| MPT | main power transformer |
| mph | miles per hour |
| MR | maintenance rule |
| Mrem | millirem |
| MSLB | main steam-line break |
| MSO | multiple spurious operations |
| MW | megawatt |
| | |
| N2 | nitrogen gas |
| NACE | National Association of Corrosion Engineers |
| NCDC | National Climatic Data Center |
| NCS | nuclear criticality safety |
| NDE | nondestructive examination |

| | |
|---|---|
| NDT | nil ductility temperature |
| NEI | Nuclear Energy Institute |
| NFPA | National Fire Protection Association |
| NFSV | new fuel storage vault |
| NOAA | National Oceanic and Atmospheric Administration |
| NOC | Nuclear Operations Center |
| NPSH | net positive suction head |
| NQA | nuclear quality assurance |
| NRC | Nuclear Regulatory Commission |
| NRCS | Natural Resources Conservation Service |
| NRO | Office of New Reactors |
| NRR | Office of Nuclear Reactor Regulation |
| NS | non-seismic |
| NTTF | Fukushima Near-Term Task Force |
| NUREG | NRC technical report designation (Nuclear Regulatory Commission) |
| NUREG/CR | NUREG contractor report |
| NWS | National Weather Service |
| | |
| OBE | operating-basis earthquake |
| ODCM | offsite dose calculation manual |
| OE | Operational Experience |
| OEM | original equipment manufacturer |
| OGS | offgas system |
| OM | Operation and Maintenance Code |
| OPC | open phase condition |
| ORE | occupational radiation exposure |
| ORO | offsite response organization |
| OSC | operational support center |
| | |
| P&ID | piping and instrumentation diagram |
| P/T | pressure/temperature |
| PA | protected area |
| PAG | Protective Action Guide |
| PAM | post-accident monitoring |
| PAR | protective action recommendation |
| PAS | post-accident sampling |
| PASS | post-accident sampling system |
| PAT | power ascension test |
| PMF | probable maximum flood |
| PMH | probable maximum hurricane |
| POV | power-operated valve |
| ppb | parts per billion |
| PRA | probabilistic risk assessment |
| PRMS | process radiation monitoring system |
| psf | pounds per square-foot |
| PSI | preservice inspection |
| psi | pounds per square inch |
| psia | pounds per square-inch absolute |
| psig | pound per square inch gauge |
| | |
| QA | quality assurance |

| | |
|---|---|
| QAP | quality assurance program |
| QAPD | quality assurance program description |
| RAI | request for additional information |
| RAP | reliability assurance program |
| RAT | reserve auxiliary transformer (alternate power source) |
| RB | reactor building |
| RCIC | reactor core isolation cooling system |
| RCPB | reactor coolant pressure boundary |
| RCS | reactor coolant system |
| rem | roentgen equivalent man (a unit of radiation dose) |
| RG | regulatory guide |
| RHR | residual heat removal system |
| RIS | regulatory issue summary |
| RMU | remote multiplexing units |
| RMS | radiation monitoring system |
| RP | radiation protection |
| RPP | radiation protection plan |
| RPV | reactor pressure vessel |
| RSS | remote shutdown panel |
| RTNSS | regulatory treatment of nonsafety systems |
| RV | reactor vessel |
| RVSP | reactor vessel (materials) surveillance program |
| RWB | radwaste building |
| RWCU | reactor water cleanup |
| RWCS | reactor water cleanup systems |
| RWMS | radioactive waste management systems |
| s | second |
| SAT | systematic approach to training |
| S/B | service building |
| SBO | station blackout |
| SCC | stress corrosion cracking |
| SDC | shutdown cooling |
| SDG | standby diesel generator |
| SDM | shutdown margin |
| SE | safety evaluation |
| SEC | Securities and Exchange Commission |
| SER | safety evaluation report |
| SFP | spent fuel pool |
| SFPC | spent fuel pool cooling |
| SGI | safeguards information |
| SLC | standby liquid control |
| SLCS | standby liquid control system |
| SPCS | suppression pool clean-up system |
| SR | surveillance requirement |
| SRI | select rod insert |
| SRM | staff requirements memorandum |
| SRO | senior reactor operator |
| SRP | Standard Review Plan |
| SRVs | reactor safety relief valves |
| SSC | structure, system, and component |

| | |
|---|---|
| SSCs | structures, systems and components |
| SSE | safe-shutdown earthquake |
| Std | Standard |
| STP | South Texas Project |
| STS | standard technical specifications |
| SUNSI | Sensitive Unclassified Non-Safeguards Information |
| Sv | Sievert |
| SWMS | solid waste management system |
| TAF | top of active fuel |
| T/B | turbine building |
| TCCWS | turbine component cooling water system |
| TEDE | total effective dose equivalent |
| TCD | thermal conductivity degradation |
| TLD | thermoluminescent dosimeter |
| TMI | Three Mile Island |
| TR | technical report |
| TS | technical specifications |
| TSC | technical support center |
| UAT | unit auxiliary transformer |
| UHS | ultimate heat sink |
| UPC | unbalanced phase condition |
| UPS | uninterruptible power supply |
| URG | utility resolution guidance |
| US | United States |
| USI | unresolved safety issues |
| USACE | U.S. Army Corps of Engineers |
| USCG | United Sates Coast Guard |
| USGS | United States Geological Survey |
| USI | unresolved safety issue |
| USNRC | United States Nuclear Regulatory Commission |
| V&V | verification and validation |
| Vac | volt alternating current |
| Vdc | volt direct current |

# 1 INTRODUCTION AND GENERAL DISCUSSION

## 1.1 Introduction

In a letter dated December 7, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110040176), GE-Hitachi Nuclear Energy (GEH or the applicant) submitted a Design Certification (DC) renewal application for the United States Advanced Boiling Water Reactor (ABWR) pursuant to the requirements of Subpart B, “Standard Design Certifications,” of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”

This report supplements the final safety evaluation report (FSER) for the ABWR standard plant design. The U.S. Nuclear Regulatory Commission (NRC) staff issued the FSER as NUREG–1503, “Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design,” in July 1994 and NUREG–1503, Supplement 1 in May 1997, to document the NRC staff's review of the ABWR. This report, Supplement 2 to NUREG–1503, documents the NRC staff's review of GEH’s application to renew the ABWR DC. Except as modified by this supplement to the FSER, the findings made in NUREG–1503 and its Supplement 1 remain in full effect. Each section of Supplement 2 is numbered and titled the same as the section of the FSER that is being updated, where applicable. The discussions and staff findings in this supplement are supplementary to, but not in lieu of, the discussions in the original FSER, unless otherwise noted.

GEH submitted the ABWR DC renewal application under Subpart B of 10 CFR Part 52. GEH’s renewal application includes the ABWR Design Control Document (DCD) and an environmental report.

Review Criteria

The following Commission regulations specify requirements for DC renewals:

1. 10 CFR § 52.57(a) states, in part, that an application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application.

2. 10 CFR § 52.59(a) states that the Commission shall issue a rule granting the renewal if the design, either as originally certified or as modified during the rulemaking on the renewal, complies with the Atomic Energy Act and the Commission’s regulations applicable and in effect at the time the certification was issued, provided, however, that the first time the Commission issues a rule granting the renewal for a standard DC in effect on July 13, 2009, the Commission shall, in addition, find that the renewed design complies with the applicable requirements of 10 CFR § 50.150, “Aircraft impact assessment.”

3. 10 CFR § 52.59(b) states that the Commission may impose other requirements if it determines that:

   a. They are necessary for adequate protection to public health and safety or common defense and security;

   b. They are necessary for compliance with the Commission's regulations and orders applicable and in effect at the time the certification was issued; or

   c. There is a substantial increase in overall protection of the public health and safety or the common defense and security to be derived from the new requirements, and the direct and indirect costs of implementing those requirements are justified in view of this increased protection.

4. 10 CFR § 52.59(c) states that the applicant for renewal may request an amendment to the DC. Section 52.59(c) also states that the Commission shall grant the amendment request if it determines that the amendment will comply with the Atomic Energy Act and the Commission's regulations in effect at the time of renewal.

In addition, while 10 CFR § 52.63(a) imposes more restrictive limits on the types of changes that may be made while a design certification rule (DCR) is in effect, 10 CFR § 52.59(c) allows the ABWR DC renewal applicant greater flexibility in seeking changes to the ABWR DC. Thus, ABWR DC renewal applications that include amendments to the certified ABWR design are not required to address the criteria in 10 CFR § 52.63. For example, the renewal applicant does not need to identify specific criteria in 10 CFR § 52.63(a)(1) as the basis for proposing an amendment to the certified design. Also, because 10 CFR § 52.63(a)(3) does not apply to DC renewal, changes made to the design during renewal are not imposed on combined license applicants and holders referencing the initial certification. However, in accordance with 10 CFR § 52.59(c), if the amendment request entails such an extensive change to the DC that an essentially new standard design is being proposed, an application for a DC must be filed in accordance with Subpart B of 10 CFR Part 52.

The design basis for the ABWR DC and DC renewal, with the exception of those design amendments proposed by the applicant in accordance with 10 CFR § 52.59(c), is based on the regulations in affect at the time of certification. While some of these regulations were specific to DCs under 10 CFR Part 52 (e.g., 10 CFR § 52.47(1)(1)(iii)-(ix) (1997)), most fell under 10 CFR § 52.47(a)(1)(i) (1997), which required that the DC application contain "[t]he technical information which is required of applicants for construction permits and operating licenses by 10 CFR part 20, part 50 and its appendices, and parts 73 and 100, and which is technically relevant to the design and not site-specific." Similarly, 10 CFR § 52.47(a)(1)(ii) (1997) required the DC application to demonstrate "compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR § 50.34(f)." The requirements referenced by 10 CFR § 52.47(a)(1)(i)-(ii) that are relevant to the ABWR are discussed in the FSER for the original certification and, as applicable, in this supplement.

<u>Review Approach</u>

Based on the regulations cited in the Review Criteria above, the NRC staff's safety review focused on ensuring that the design, as modified, is consistent with 10 CFR

§ 52.59(a) and that any amendments comply with 10 CFR § 52.59(c). The NRC staff review also focused on ensuring that the entire ABWR DCD (i.e., the version of the ABWR DCD last approved for incorporation by reference) was updated under 10 CFR § 52.57(a). Updates include clarifications consistent with the original understanding of the design information, and corrections of errors, typos, and defects (as defined in 10 CFR Part 21, “Reporting of Defects and Noncompliance”). In addition, the ABWR DCD was updated to include the information necessary to demonstrate the technical qualification of the applicant because GEH is not the original applicant for the ABWR DC. General Electric Nuclear Energy (GENE) was the original applicant for the ABWR DC that became effective on June 11, 1997. In 2007, General Electric Company and Hitachi formed an alliance, and GEH became the entity to retain the ABWR design information of predecessor to GENE. GEH has been involved in the design and development of commercial nuclear power plants, reactor plant designs and nuclear fuel for boiling-water reactors. Therefore, based on the above, GEH is technically qualified to supply the design.

To support the initial certification of the ABWR, the NRC determined that the design was safe and complied with NRC requirements. Therefore, consistent with the Commission’s direction in the May 12, 1997, final rule for the original certification of the ABWR (62 FR 25800, 25804-05), the NRC staff did not perform a *de novo* review of GEH’s renewal application. Instead, the staff’s review conformed to the Commission’s expectation that “the review focus would be on changes to the design that are proposed by the applicant and insights from relevant operating experience with the certified design or other designs, or other material new information arising after the NRC staff's review of the DC.”

For those sections of the ABWR DCD that the applicant did not propose to change, the NRC staff evaluated whether the staff’s findings in NUREG–1503 and NUREG–1503, Supplement 1 are still valid. This evaluation was based on the consideration of the following types of information:

1. Errors (including typographical errors) and defects (as defined in 10 CFR Part 21) that should result in corrections to the DCD;

2. Material new information with respect to technical resolutions to high and medium priority unresolved safety issues (USIs) and generic safety issues (GSIs) addressed in the original ABWR DCR;

3. New USIs and GSIs created or identified since the ABWR design was certified;

4. New generic letters and bulletins issued after the ABWR design was certified;

5. Any relevant domestic and international operating experience that has been documented since the ABWR design was certified; and

6. Any other new, material information of which the staff is aware that invalidates the staff’s findings in NUREG–1503 and NUREG–1503, Supplement 1.

The applicant provided information to support the staff’s consideration described above in letters dated August 31, 2016 (ADAMS Accession No. ML16244A122), November 17, 2016 (ADAMS Accession No. ML16323A003), and December 13, 2016 (ML16348A096). In determining whether the staff’s findings in support of the original certification are still valid, the staff sought additional information from the applicant on some issues.  In some cases, the applicant proposed design changes to address the staff's questions, and in other cases the staff determined that no change was necessary.  For those sections that have not changed in the ABWR DCD, the staff did not identify any new information of the type described above that would invalidate the findings in NUREG–1503 and NUREG–1503, Supplement 1.  Therefore, the staff concludes that the unchanged sections of the DCD continue to comply with the Atomic Energy Act and the Commission's regulations applicable and in effect at the time the certification was issued per 10 CFR § 52.59(a). For those sections that have changed in the ABWR DCD as a result of the consideration described above, the staff's supplemental FSER includes a discussion of the specific matter associated with the design change.

The staff considers design changes to fall in three categories.  These categories are: modifications, renewal backfits, and amendments.  Therefore, the staff evaluated design changes as follows:

1. *Modifications* to the certified design are those changes made pursuant to the requirement to update the application in accordance with § 52.57(a) (e.g., clarifications consistent with the original understanding of the design information, changes to correct known errors, typos, or defects as defined in 10 CFR Part 21) or to meet the standards for renewal in § 52.59(a).[2]  Modifications include proposed changes in response to NRC staff concerns on whether the § 52.59(a) standards are met.  As required by § 52.59(a), modifications must comply with the Atomic Energy Act of 1954, as amended (AEA), and the Commission's regulations applicable and in effect at the time the certification was originally issued with the exception of those changes proposed by the DC renewal applicant to comply with 10 CFR § 50.150.

2. *Renewal backfits* to the certified design are those changes that are necessary to comply with additional requirements imposed by the NRC through application of the criteria in § 52.59(b).  The NRC staff is responsible for justifying renewal backfits under this provision.

3. *Amendments* to the certified design are those changes proposed by the DC renewal applicant in accordance with § 52.59(c).  Amendments must comply with the AEA and the Commission's regulations applicable and in effect at the time of renewal.  If the amendment request entails such an extensive change to the certified design that an essentially new standard design is being proposed, a new DC application must be submitted.

Renewal backfits are changes imposed by the NRC, while modifications and amendments are changes proposed by the applicant.  If a design change is made to satisfy the updating requirement in 10 CFR § 52.57(a) or to meet the standards in 10 CFR § 52.59(a), then the change is a modification and must comply with the regulations applicable and in effect at the time the certification was issued.  Otherwise the change is an amendment and must satisfy the regulations in effect at the time of renewal.

[2] The term "modification" derives from 10 CFR § 52.59(a), which refers to the "design, either as originally certified or as *modified* during the rulemaking on the renewal" (emphasis added).

This supplement is issued by the Division of New and Renewed Licenses in the Office of Nuclear Reactor Regulation, NRC. The NRC's project manager for the review of GEH's ABWR DC renewal application is James Shea. He may be reached by calling 301-415-1388, or by writing to him at the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. The ABWR design documentation and all revisions are available for public inspection at the NRC's Public Document Room and in ADAMS.[3] Through the NRC public website (https://www.nrc.gov/), the public can gain electronic access to ADAMS, which provides text and image files of NRC's public documents. The ABWR FSER (NUREG–1503 and NUREG–1503, Supplement 1) as well as this supplement are also available for public inspection in ADAMS and the ABWR DC Renewal public web-site (https://www.nrc.gov/reactors/new-reactors/design-cert/renewal-abwr-ge-hitachi.html#safety).

## 1.5 Summary of Principal Review Matters

By letter dated December 7, 2010 (ADAMS Accession No. ML110040176), GEH submitted an application to renew the ABWR DC. ABWR DCD, Revision 5 was included in the applicant's December 7, 2010 submittal. The NRC staff reviewed the application and, in a letter dated July 20, 2012 (ADAMS Accession No. ML12125A385), identified proposed changes that were considered to be regulatory improvements or changes that could meet the criteria in 10 CFR § 52.59(b). These suggested changes by the staff for GEH consideration included recommendations contained in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A111), addressing Recommendations 4.2, 7.1 and 9.3 from the Fukushima Near-Term Task Force Report, and SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated July 12, 2011 (ADAMS Accession No. ML11186A950).

Subsequent to the staff's 2012 letter to GEH, the NRC staff issued several requests for additional information (RAIs) to identity additional items or clarify the items communicated in the 2012 letter. By letter dated February 19, 2016 (ADAMS Accession No. ML16081A268), the applicant submitted ABWR DCD, Revision 6, the first revision of its application to incorporate changes to the ABWR DCD that were previously communicated to the NRC via letters responding to the 2012 staff letter and to the staff's RAIs. In addition, this revision transmitted corrections of typographical mistakes that were uncovered during document development and other required formatting changes. These corrections represent non-substantive changes that are editorial in nature. The NRC staff reviewed these typographical changes and determined that they do not affect the staff's findings in the FSER for initial certification and are acceptable.

For the staff-suggested changes in Items 14, 15, 16, 21, 24, and 25 in the 2012 staff letter, the applicant informed the NRC staff that changes will not be made to the ABWR

[3] ADAMS is the NRC's information system that provides access to all image and text documents that the NRC has made public since November 1, 1999, as well as bibliographic records (some with abstracts and full text) that the NRC made public before November 1999. Documents available to the public may be accessed via the Internet at https://www.nrc.gov/reading-rm/adams.html. Documents may also be viewed by visiting the NRC's Public Document Room at One White Flint North, 11555 Rockville Pike, Rockville, Maryland. Telephone assistance for using web-based ADAMS is available at (800) 397-4209 between 8:30 a.m. and 4:15 p.m. eastern time, Monday through Friday, except Federal holidays.

DCD through the renewal application. In letters dated September 25, 2015, August 14, 2015, June 03, 2016, and September 11, 2015 (ADAMS Accession Nos. ML15271A171, ML15226A416, ML16155A025, and ML15258A666), GEH submitted justifications explaining that the original ABWR DC contains sufficient information with respect to these items. These items relate to: (1) probabilistic risk assessment, (2) instrumentation and controls system design, (3) inspections, tests, analyses, and acceptance criteria, and (4) human factors engineering.

In a letter dated February 2, 2018 (ADAMS Accession No. ML17097A470), the NRC staff provided its review with respect to these items. In summary, the staff determined that Items 14, 15, 16, 21, 24, and 25 are not necessary for compliance with the applicable regulations in effect at initial certification and, therefore, are also not necessary for reasonable assurance of adequate protection of the public health and safety. For this reason, incorporation of these suggested improvements is not necessary to support the findings required by 10 CFR § 52.59(a) to renew the DC. The staff has also decided that further evaluating these improvements through the 10 CFR § 52.59(b) process is not warranted.

The remaining items identified in the 2012 staff letter, as well as the RAIs issued by the NRC staff, resulted in the applicant proposing changes to the ABWR DCD to address the staff's concerns. Therefore, the NRC staff did not need to impose any backfits during the renewal review.

Following the submittal of the ABWR DCD, Revision 6, the applicant provided additional or alternative information in ABWR DCD, Revision 7, submitted December 2019 (ADAMS Accession No. ML20007E371), which incorporated the appropriate changes described in the applicant's responses and letters submitted after ABWR DCD, Revision 6. Therefore, all the Confirmatory Items from the staff's advanced safety evaluation with no open items for the ABWR DC renewal are resolved and closed.

The table below identifies the supplemental FSER sections with the staff's evaluations of the ABWR DC changes contained in the renewal application and identifies whether the changes are modifications or amendments. The amendments are limited in nature, and do not entail such an extensive change to the certified design that an essentially new standard design is being proposed.

**TABLE 1-1 Design Change Categories**

| SER Section | Amendment/Modification |
|---|---|
| Section 2.3, Meteorology | Modification |
| Section 2.5, Geological, Seismological and Geotechnical Engineering | Modification |
| Section 2.6.2, Water Level (Flood) Design Site Parameters | Modification |

| SER Section | Amendment/Modification |
|---|---|
| Section 2.6.8, ABWR Site Acceptability | Modification |
| Sections 3.2.3, Safety Classifications | Amendment |
| Section 3.3, Wind and Tornado Loadings | Modification |
| Section 3.5.1.4, Missiles Generated by Natural Phenomena | Modification |
| Section 3.7.3, Seismic Subsystem Analysis | Modification |
| Section 4.2, Fuel System Design | Modification |
| Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection | Amendment |
| Section 5.4.7, Residual Heat Removal System | Amendment |
| Section 5.4.7.1.1.10, ACIWA | Amendment |
| Section 5.4.8, Reactor Water Cleanup System | Amendment |
| Section 6.2.1.3, Short-Term Pressure Response | Amendment |
| Section 6.2.1.6, Suppression Pool Dynamic Loads | Modification |
| Section 6.2.1.9, Containment Debris Protection for ECCS Strainers | Amendment |
| Section 7.4.1.4.4, Shutdown Panel | Amendment |
| Section 7.5.2.1, Post Accident Monitoring System | Amendment |

| SER Section | Amendment/Modification |
|---|---|
| Section 7.7.1.2.1, Control Rod Ganged Withdrawal Sequence Restrictions | Modification |
| Section 8.2.5, NRC Bulletin 2012-01: Design Vulnerability In Electric Power System | Modification |
| Section 8.3.4.4, Isolation Between Class 1E Buses and Loads Designated as Non-Class 1E | Amendment |
| Section 9.1.1, New Fuel Storage | Amendment |
| Section 9.1.2.1, New and Spent Fuel Storage | Modification |
| Section 9.1.2.2, Fuel Racks | Amendment |
| Section 9.1.3, Fuel Pool Cooling and Cleanup System | Amendment |
| Section 9.1.4, Light Load Handling System (Related to Refueling) | Amendment |
| Section 9.1.5, Overhead Heavy Load Handling Systems | Amendment |
| Section 9.5.1, Fire Protection | Modification |
| Section 11.4, Solid Waste Management System | Modification |
| Section 12.2, Radiation Sources | Modification |
| Section 12.3, Radiation Protection Design Features | Amendment |
| Section 13.3, Emergency Planning Technical Support Center Changes | Modification |
| Section 13.3, Emergency Planning Communications & Staffing Enhancements | Amendment |

| SER Section | Amendment/Modification |
|---|---|
| Section 13.5, Plant Procedures | Amendment |
| Section 14.3.2.3.6, Structural Task Group Review | Modification |
| Section 16, Technical Specifications | Amendment |
| Section 19.2.3.3.4, ABWR Containment Vent Design | Modification |
| Section 19.5, Aircraft Impact Assessment | Modification |
| Section 22.0, Requirements Resulting from Fukushima Near Term Task Force Recommendations | Amendment |

# 2 SITE CHARACTERISTICS

## 2.3 <u>Meteorology</u>

### 2.3.1 Regional Climatology

#### *2.3.1.1 Regulatory Criteria*

In accordance with NRC regulations, nuclear plants must be designed so that they remain in a safe condition under extreme meteorological events, including those that could result in the most extreme wind events (tornadoes and hurricanes) that could reasonably be predicted to occur. The applicant added hurricane wind speed and hurricane missile spectra to the list of site parameter values presented in DCD Tier 1, Section 5.0, and DCD Tier 2, Section 2.0, of the GEH ABWR DCD, Revision 7. A combined license (COL) applicant that references the GEH ABWR DC will assess whether the actual site characteristics fall within the site parameters specified for the ABWR design.

The applicant made changes to the ABWR DCD, Revision 7, to provide criteria for a COL applicant to determine whether an ABWR located at a particular site is appropriately protected against the effects of hurricane winds and missiles. In September 2014, the staff issued request for additional information (RAI) 02-1 (ADAMS Accession No. ML14267A352), raising concerns about compliance with GDC 2 (1997) and 4 (1997) for hurricane loads and hurricane-generated missiles. In response, the applicant added information to DCD Tier 1, Section 5.0 and Tier 2, Section 2.0. Since the applicant's changes were in response to the staff's concerns regarding compliance with regulations in effect at initial certification, these changes are "modifications," as described in Chapter 1 of this FSER supplement, and the staff will therefore evaluate them using the regulations applicable and in effect at the time of initial certification.

The applicable regulatory requirements for evaluating the proposed changes are as follows:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC) 2, "Design Bases for Protection Against Natural Phenomena," (1997), requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety function.
- GDC 4, "Environmental and Dynamic Effects Design Bases," (1997), requires, in part, that SSCs important to safety to be appropriately protected against dynamic effects, including the effects of missiles that may result from equipment failures and from events and conditions outside the nuclear power unit.
- 10 CFR § 52.47(a)(1)(iii) (1997) states that DC applications must include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters.

Additional information on the staff's review of DC renewal applications with respect to hurricane wind and hurricane missile site parameters can be found in DC/COL-ISG-024, "Implementation of Regulatory Guide 1.221 on Design-Basis Hurricane and Hurricane Missiles," issued May 2013.

### 2.3.1.2 Summary of Technical Information

In this supplemental FSER section the staff evaluates the proposed hurricane wind site parameters. Sections 3.5.1.4 and 3.3 of this supplemental FSER provide the staff's evaluation of the missiles generated by hurricane winds and the resulting extreme wind loadings on structures important to safety, respectively.

In the applicant's response to RAI 2.0-1 dated November 19, 2014 (ADAMS Accession No. ML14324A082), GEH provided: (1) DCD Tier 1 site parameters related to hurricane maximum wind speed, maximum pressure drop, and missile spectra, and (2) DCD Tier 2 site parameters related to hurricane maximum wind speed, maximum rotational speed, translational velocity, radius, maximum pressure drop, and missile spectra.

Subsequently, GEH revised its RAI 02-1 response in Supplement 1 dated June 26, 2015 (ADAMS Accession No. ML15177A038), by eliminating the following site parameters for the hurricane: (1) maximum pressure drop from the list of DCD Tier 1 site parameters, and (2) maximum rotational speed, translational velocity, radius, and maximum pressure drop from the list of DCD Tier 2 site parameters. As discussed in DC/COL-ISG-024, the load from the hurricane atmospheric pressure change is assumed to be small. The rate of pressure change at a specific location from the passage of a hurricane is slow compared to the passage of a tornado because the large pressure drop within a hurricane occurs over a distance of several miles, whereas the large pressure drop within a tornado occurs over a few hundred feet. Consequently, the staff evaluated these parameters and concludes that listing hurricane maximum rotational speed, translational velocity, radius, and maximum pressure drop as site parameters is not necessary as these site parameters are used to determine the rate of hurricane atmospheric pressure drop which is assumed to be small.

GEH further revised its RAI 02-1 response in Supplement 5, dated April 13, 2017, by proposing to modify the DCD to indicate that the severe wind and extreme hurricane wind site parameter values are fastest-mile values, consistent with the wind loading methodology at the time of initial certification as presented in American National Standards Institute/American Society of Civil Engineers (ASCE) Section 7-88, 1990, "Minimum Design Loads for Buildings and Other Structures." Additionally, as part of its response to RAI 02-1, the applicant proposed changes to the DCD that state the extreme maximum tornado wind speed site parameter value is a fastest quarter mile value that is consistent with the wind loading methodology at the time of initial certification as presented in the NRC approved Bechtel Topical Report BC-TOP-3-A, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," Revision 3, issued August 1974 (ADAMS Accession No. ML14093A218). The applicant also provided the corresponding equivalent 3-second gust values in DCD Tier 1, Table 5.0 and DCD Tier 2, Table 2.0-1.

The staff reviewed the hurricane wind speed contour maps in RG 1.221, Revision 0, and concluded that, except for certain locations along the Gulf and the Atlantic coasts, a design-basis hurricane 3-second wind speed site parameter value of 286.5 km/h (178 mph) is bounding. Because the proposed ABWR design-basis hurricane wind speed site parameter value bounds a reasonable number of potential COL sites, the staff finds the proposed site parameter value acceptable. If the design-basis hurricane wind speed site parameter value defined in the ABWR DCD does not bound a particular site, the COL applicant referencing the design will need to request an exemption from the Tier 1 site parameter as part of its application and submit analyses to demonstrate that the site-specific hurricane wind speed value does not exceed the capability of the design.

The applicant included the changes described in Supplement 5 of the response to RAI 02-1, dated April 13, 2017, in the ABWR DCD, Revision 7. Therefore, Confirmatory Item 2.3-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*2.3.1.3 Technical Evaluation*

In this supplemental FSER section the staff evaluates the proposed hurricane wind site parameters. Sections 3.5.1.4 and 3.3 of this supplemental FSER provide the staff's evaluation of the missiles generated by hurricane winds and the resulting extreme wind loadings on structures important to safety, respectively.

In the applicant's response to RAI 2.0-1 dated November 19, 2014 (ADAMS Accession No. ML14324A082), GEH provided: (1) DCD Tier 1 site parameters related to hurricane maximum wind speed, maximum pressure drop, and missile spectra, and (2) DCD Tier 2 site parameters related to hurricane maximum wind speed, maximum rotational speed, translational velocity, radius, maximum pressure drop, and missile spectra.

Subsequently, GEH revised its RAI 02-1 response in Supplement 1 dated June 26, 2015 (ADAMS Accession No. ML15177A038), by eliminating the following site parameters for the hurricane: (1) maximum pressure drop from the list of DCD Tier 1 site parameters, and (2) maximum rotational speed, translational velocity, radius, and maximum pressure drop from the list of DCD Tier 2 site parameters. As discussed in DC/COL-ISG-024, the load from the hurricane atmospheric pressure change is assumed to be small. The rate of pressure change at a specific location from the passage of a hurricane is slow compared to the passage of a tornado because the large pressure drop within a hurricane occurs over a distance of several miles, whereas the large pressure drop within a tornado occurs over a few hundred feet. Consequently, the staff evaluated these parameters and concludes that listing hurricane maximum rotational speed, translational velocity, radius, and maximum pressure drop as site parameters is not necessary as these site parameters are used to determine the rate of hurricane atmospheric pressure drop which is assumed to be small.

GEH further revised its RAI 02-1 response in Supplement 5, dated April 13, 2017, by proposing to modify the DCD to indicate that the severe wind and extreme hurricane wind site parameter values are fastest-mile values, consistent with the wind loading methodology at the time of initial certification as presented in American National Standards Institute/American Society of Civil Engineers (ASCE) Section 7-88, 1990, "Minimum Design Loads for Buildings and Other Structures." Additionally, as part of its response to RAI 02-1, the applicant proposed changes to the DCD that state the extreme maximum tornado wind speed site parameter value is a fastest quarter mile value that is consistent with the wind loading methodology at the time of initial certification as presented in the NRC approved Bechtel Topical Report BC-TOP-3-A, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," Revision 3, issued August 1974 (ADAMS Accession No. ML14093A218). The applicant also provided the corresponding equivalent 3-second gust values in DCD Tier 1, Table 5.0 and DCD Tier 2, Table 2.0-1.

The staff reviewed the hurricane wind speed contour maps in RG 1.221, Revision 0, and concluded that, except for certain locations along the Gulf and the Atlantic coasts, a design-basis hurricane 3-second wind speed site parameter value of 286.5 km/h (178 mph) is bounding. Because the proposed ABWR design-basis hurricane wind speed site parameter value bounds a reasonable number of potential COL sites, the staff finds the proposed site parameter value acceptable. If the design-basis hurricane wind speed site parameter value defined in the ABWR DCD does not bound a particular site, the COL applicant referencing the design will need to request an exemption from the Tier 1 site parameter as part of its application and submit analyses to demonstrate that the site-specific hurricane wind speed value does not exceed the capability of the design.

The applicant included the changes described in Supplement 5 of the response to RAI 02-1, dated April 13, 2017, in the ABWR DCD, Revision 7. Therefore, Confirmatory Item 2.3-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *2.3.1.4 Conclusion*

Based on the evaluation provided in this FSER section supplement, the staff concludes that the changes to add a maximum hurricane wind speed as a DCD Tier 1 and DCD Tier 2 site parameter to the ABWR DCD, Revision 7, are acceptable and do not alter the safety findings made in NUREG–1503 and meet the applicable regulations in effect at initial certification, including the requirements of GDC 2 (1997) and GDC 4 (1997). The staff’s review concludes that the applicant added an appropriate design-basis hurricane wind speed site parameter for the GEH ABWR, and therefore complies with 10 CFR § 52.47(a)(1)(iii) (1997). This is consistent with the guidance in RG 1.221, Revision 0, for design-basis hurricane wind speeds for nuclear power plants and therefore is acceptable.

## 2.5 Geological, Seismological, and Geotechnical Engineering

### *2.5.1 Regulatory Criteria*

The GEH ABWR design is certified for plants founded on soil deposits up to 91.5 meters (300 feet), in addition to rock sites. Therefore, there is a potential that larger differential settlements may occur for a deep soil site due to the geologic variation of subsurface materials and non-uniform loading distribution. The applicant added dynamic bearing capacity and differential site parameters to the ABWR DCD in order to ensure that the soil under the foundation and the foundation itself will be able to withstand the foundation dynamic pressure resulting from the combination of all possible loadings. These parameters are ABWR DCD clarifications that demonstrate compliance to applicable regulations at the time of original certification. Therefore, this design change is a “modification,” as that term is defined in Chapter 1 of this supplement and will correspondingly be evaluated using the regulations applicable and in effect at the initial ABWR certification.

The applicable regulatory requirements for evaluating the ABWR DCD modifications related to geology, seismology, and geotechnical engineering design parameters are as follows:

- 10 CFR § 52.47(a)(1)(iii) (1997) requires DC applicants to provide postulated site parameters, and an analysis and evaluation of the design in terms of such parameters.

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 2, "Design Bases for Protection Against Natural Phenomena," (1997) with respect to structures, systems, and components (SSC) important-to-safety being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- 10 CFR Part 100, Appendix A,[4] Section V.(d) (1997), requires that each applicant determine whether soil instability will result from vibratory ground motion associated with the safe-shutdown earthquake.

### *2.5.2 Summary of Technical Information*

In the ABWR DCD, Revision 7, the applicant presented design parameters and associated combined license (COL) Information Items related to geology, seismology, and geotechnical engineering in DCD Tier 1, Section 5.0, "Site Parameters"; DCD Tier 2, Section 2.0, "Site Characteristics"; and DCD Tier 2, Section 2.3, "COL License Information".

The applicant described seismic design parameters including safe-shutdown earthquake ground motion, bearing capacity, and settlement in DCD Tier 2, Section 2.3.1.2, "Seismic Design Parameters".

COL requirements for basic geologic and seismic information, vibratory ground motion, surface faulting, stability of subsurface material and foundation, site and facilities, field investigations, laboratory investigations, subsurface conditions, excavation and backfilling for foundation construction, effect of ground water, liquefaction potential, response of soil and rock to dynamic loading, minimum soil bearing capacity, earth pressures, soil properties for seismic analysis of buried pipes, static and dynamic stability of facilities, subsurface instrumentation, stability of slopes, and embankments and dams are described in DCD Tier 2, Section 2.3.2.21 through Section 2.3.2.39 respectively.

[4] The requirements of 10 CFR Part 100, Appendix A, apply here because 10 CFR Part 100, Subpart B, applies only to applications submitted on or after January 10, 1997.

The applicant added additional information that is related to geology, seismology, and geotechnical engineering design parameters to the ABWR DCD, Revision 7. The additional information (represented below with italicized text) was submitted originally in ABWR DCD, Revision 6 as follows:

**DCD Tier 1, 5.0 Site Parameters**

**DCD Tier 2, Table 5.0 ABWR Site Parameters**

*Minimum Dynamic Bearing Capacity: 2,700 kPa [392 psi]*
*Maximum Settlement(9): 75mm [2.95 in.]*
*Maximum Foundation Angular Distortion: 1/750(10)*
*Note: (9) Settlement is long term (post construction) value.*
*(10) Angular distortion is defined as the slope between two adjacent columns. Angular distortion is long term (post construction) value.*

**DCD Tier 2, 2.0 Site Characteristics**

**DCD Tier 2, Table 2.0-1 Envelope of ABWR Standard Plant Site Design Parameters**

*– Maximum Dynamic Bearing Capacity: 2,700 kPa [392 psi]*
*– Maximum Settlement: 75mm [2.95 in.] †††*
*– Maximum Foundation Angular Distortion: 1/750 ‡‡‡*
*Note: ††† Settlement is long term (post construction) value.*
*‡‡‡ Angular distortion is defined as the slope between two adjacent columns. Angular distortion is long term (post construction) value.*

**DCD Tier 2, 2.3 COL License Information**

**DCD Tier 2, Section 2.3.1.2 Seismic Design Parameters**

(2) Bearing Capacity

The site soil static bearing capacity at the foundation level of the reactor and control building is 718.20 kPa [104 psi] minimum. *The maximum static bearing demand is compared with the site-specific allowable static bearing pressure, which is obtained by dividing the ultimate soil bearing capacity by a factor of safety appropriate for the design load combination. The maximum dynamic bearing demand is compared with the site-specific allowable dynamic bearing pressure, which is obtained by dividing the ultimate soil bearing capacity by a factor of safety appropriate for the design load combination.*
*The site soil dynamic bearing capacity at the foundation level of the reactor and control building is 2,700 kPa [392 psi] minimum.*

*(3) Settlement*

*The maximum settlement of the reactor and control building foundations is 75mm [2.95 in.]. The maximum angular distortion of the reactor and control building is 1/750.*

### 2.5.3 *Technical Evaluation*

The staff reviewed the ABWR DC renewal modifications related to geology, seismology, and geotechnical engineering design parameters and the associated sections in NUREG–1503 and its Supplement 1, the FSER for the original DC. The staff's technical evaluation focused on the technical basis of the design parameters and the adequacy of associated COL Information Items.

The dynamic bearing capacity and differential settlement site parameters are important design requirements to ensure the stability of foundation and structure for a nuclear power plant, in a RAI Question 02.05.04-1, dated June 9, 2015 (ADAMS Accession No. ML15160A421), the staff asked the applicant to add these site parameters to the DCD and to provide details on how the dynamic bearing capacity and differential settlement site parameters were determined, including the model(s), assumptions and input parameters used in analyses and calculations, and justifications for site parameter value determinations.

In the applicant's response on July 24, 2015 (ADAMS Accession No. ML15209A561), November 13, 2015 (ADAMS Accession No. ML15317A092) and May 31, 2016 (ADAMS Accession No. ML16152A512), GEH provided additional site parameters with detailed descriptions and justifications.

The applicant incorporated all design changes from its RAI 02.05.04-1 responses in the ABWR DCD, Revision 7. This includes clarifying DCD Tier 2, Table 2.0-1, to reflect that the 2,700 kilopascals (kPa) (392 pounds per square inch (psi)) value represents the minimum dynamic bearing capacity site parameter. As discussed below, Confirmatory Item 02.05.04-1 from the staff's advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

The applicant stated that since the site parameter for minimum static bearing capacity in the originally certified ABWR DCD was determined by adding a margin factor to the calculated maximum static foundation pressure value, the same approach was used in the determination of the minimum dynamic bearing capacity site parameter. The calculated maximum dynamic bearing pressure for the ABWR reactor building (the heaviest building) was 2,336 kPa (339 psi), as documented in DCD Tier 2, Section 3H.1.5.6 (unchanged from the originally certified DCD). Based on this calculation, the applicant specified the minimum dynamic bearing capacity site parameter as 2,700 kPa (392 psi) to provide some margin. The applicant further specified that the site-specific dynamic bearing capacity determined at the COL application stage should be obtained by dividing the ultimate soil bearing capacity by a factor of safety appropriate for the design load combination, which is described in its revised COL Information Item 2.3.1.2 (2) in DCD Tier 2, Section 2.3.1.2.

The staff reviewed the applicant's RAI responses and related documents. (1) the staff reviewed DCD Tier 2, Section 3H.1.5.6 and confirmed that the calculated maximum foundation bearing pressure under the combination of seismic and other loads was specified as 2,336.0 kPa (339 psi), which is the same as that in the certified ABWR DCD, Revision 4; (2) the applicant specified the minimum dynamic bearing capacity site parameter as 2,700 kPa (392 psi), which is about 15 percent higher than the calculated maximum foundation bearing pressure value; and (3) the ABWR DCD requires a factor of safety appropriate for the design load combinations to be used when determining site specific soil dynamic bearing capacity. The combination of the higher site parameter value than the calculated one and the requirement of an appropriate factor of safety to be used when determining the site-specific soil dynamic bearing capacity will provide an adequate safety margin that accounts for the variability and uncertainties of subsurface materials and dynamic/seismic loadings. The staff therefore concludes that the specified dynamic bearing capacity site parameter is adequate because it will provide a design basis for subsurface material underneath the structure foundations to withstand maximum foundation pressure generated by the structure's response to the combination of designed dynamic/seismic and dead loadings.

The applicant specified a total long term (post-construction) settlement of 75 millimeters (mm) (2.95 inches (in.)) as a site parameter based on ABWR construction experience. The staff concludes that the long term settlement limit of 75 mm (2.95 in.) is reasonable for the GEH ABWR structures because total settlements up to 125 mm (4.92 in.) can be tolerated without damage for buildings constructed on reinforced concrete mat or raft foundation according to the commonly accepted industrial guidance (e.g., engineering manual of the U.S. Army Corps of Engineers) and engineering practices.

As angular distortion, defined as the slope between two adjacent column lines, is one of the foundation differential settlement measurements that affects foundation stability, the applicant specified the maximum angular distortion limit as 1/750. The staff considers this angular distortion limit to be acceptable because the commonly accepted limits for angular distortion are in the range of 1/500 to 1/750 according to industrial guidance and practices (e.g., engineering manual of the U.S. Army Corps of Engineers;) the staff, therefore, concludes that defining the angular distortion limit at 1/750, the lower end of this range, meets the foundation stability requirement and will not have an adverse effect on structures housing equipment sensitive to differential settlement.

For other issues related to differential settlement, such as the effect of building settlement on the connection of other components to the buildings, the applicant stated that even with an aggressive 39 month construction schedule, the mechanical and electrical components would be installed at least 12 months after the completion of the foundation basemat, which allows sufficient time for the buildings to settle. The applicant also stated that because the ABWR primary containment penetrations sleeves are fixed and some component positions cannot be adjusted after its construction, the ABWR primary containment shares a common basemat with the reactor building, and openings will be left in exterior walls to allow for the installation of components after construction of the wall and these openings are made large enough to account for expected settlement. The applicant further stated that the ABWR DCD does not need to have a design value for the differential settlement between buildings because the maximum differential settlement is the same as the building's maximum settlement value. The staff considers the applicant's statement that building settlement will not affect the connection of components to the buildings is reasonable because (1) engineering practices have shown that more than 95 percent of total building settlement will occur within 12 months of construction completion for suitable nuclear power plant foundation supporting materials (e.g., well compacted granular materials;) and (2) the design and construction procedure of the wall openings for component connections will accommodate the residual long-term settlement. The staff therefore concludes that the specified allowable foundation settlement will have no adverse effect on proper component connections to the buildings. Since the ABWR primary containment shares a common basemat with the reactor building, these two buildings will have the same settlement, and the design and sequences of building construction and component connection will ensure the proper installation of components between buildings. Therefore, the staff agrees that no other differential settlement requirement, other than the angular distortion limit, is needed for the ABWR design.

Based on the above findings, the staff concludes that the applicant adequately addressed the issues related to minimum dynamic bearing capacity and settlement limit requirements, and the modifications related to geology, seismology, and geotechnical engineering design parameters will provide additional assurance of the stability and safety of the nuclear power plant structures.

The applicant provided the necessary information in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's responses to RAI 02.05.04-1. Therefore, Confirmatory Item 02.05.04-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *2.5.4 Conclusion*

Based on the review of the applicant's design modifications related to the geology, seismology, and geotechnical engineering presented in the ABWR DCD, Revision 7, and the applicant's RAI responses, the staff concludes that the applicant adequately specified additional clarification on site parameters that include minimum dynamic bearing capacity, long term settlement limits and angular distortion limit in the ABWR DCD, with associated COL Information Items. The added site parameters were determined based on NRC approved analysis procedures and/or in conformance with the commonly accepted industrial guidance and practices, which will provide additional assurance of the foundation and structure stability. The staff also concludes that the new and revised COL Information Items associated with the added site parameters adequately direct COL applicants referencing the ABWR DC renewal to meet those site parameter requirements. Therefore, the staff concludes that the design modifications related to geology, seismology, and geotechnical engineering design parameters and associated COL application requirements meet the regulatory requirements of 10 CFR § 52.47(a)(1)(iii), GDC 2, and 10 CFR Part 100, Appendix A, Section V.(d).

### 2.6.2 Water Level (Flood) Design Site Parameters

#### *2.6.2.1 Regulatory Criteria*

ABWR DCD Tier 2, Section 2.1, Revision 7, provides site parameters (including groundwater levels), that are requirements for site acceptability that must be met by combined license applicants that reference the ABWR design. DCD Tier 2, Section 2.3.2.34 provides information related to the hydrostatic groundwater pressures acting on plant safety-related facilities.

In the staff July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 4 the staff suggested consideration of the following: (1) the significance of design basis maximum groundwater level in the hydrology section and its allowable margin if any; (2) identify where this parameter is used; and (3) if feasible, set the design basis maximum groundwater level at site grade.

In a letter dated August 24, 2015 (ADAMS Accession No. ML15236A226) the applicant proposed to add a reference in DCD Tier 2, Chapter 2, as the basis for the standard plant site design parameter of “Maximum Ground Water Level” listed in DCD Tier 2, Table 2.0-1. The applicant did not change the existing groundwater level site parameter from the originally certified design, but rather clarified the basis for the site parameter. Therefore, this change is a “modification,” as that term is defined in Chapter 1 of this supplement and will correspondingly be evaluated using the regulations applicable and in effect at the initial ABWR certification. The clarification was evaluated using 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants.” (GDC) 2, “Design Bases for Protection Against Natural Phenomena,” which requires that structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as floods and high groundwater, and 10 CFR § 52.47(a)(1)(iii), which requires site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters. The acceptance criteria for site specific limits imposed on maximum groundwater level are given in DCD Tier 2, Table 2.0-1 as 61 centimeters (cm) (2.0 feet (ft)) below grade. The staff reviewed the applicants ABWR DCD, Revision 7, change related to the maximum groundwater level site parameter against the acceptance criteria of NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) Section 3.4.2, “Analysis Procedures,” Revision 2, March 2007.

#### *2.6.2.2 Summary of Technical Information*

DCD Tier 2, Section 2.0 and Table 2.0-1 was revised in ABWR DCD Revision 6. The applicant stated that the groundwater level is used in determining the at-rest soil pressure and hydrostatic pressure on buildings and below grade exterior walls. As applicable, the groundwater level is also used in determining the shear wave and compression wave velocity of soil which are used in the performance of the soil-to-structure interaction analysis. GEH added a reference to DCD Tier 2, Section 2.0 and modified a footnote in DCD Tier 2, Table 2.0-1 to reflect that the “Maximum Ground Water Level” and “Maximum Flood (or Tsunami) Level” site parameters are based on technical requirements in the Electric Power Research Institute Utility Requirements Document that have been agreed to by the industry and found acceptable by the NRC (NUREG–1242, “NRC Review of Electric Power Research Institute’s Advanced Light Water Reactor Utility Requirements Document,” issued August 1992). The applicant noted that changing the ABWR DCD groundwater level to site grade would not be possible without impacting analyses in DCD Tier 2, Appendices 3A “Seismic Soil Structure Interaction Analysis” and Appendix 3H “Design Details and Evaluation Results of Seismic Category I Structures”.

#### *2.6.2.3 Technical Evaluation*

The changes reflected in the ABWR DCD, Revision 7, described above, continue to meet established guidance and does not revise groundwater or flooding analyses previously reviewed and found acceptable by the NRC staff in NUREG–1503 and do not affect any previous staff findings of regulatory compliance or reasonable assurance of adequate protection of public health and safety related to the ABWR design. Therefore, the staff finds that the applicant’s changes to DCD Tier 2, Table 2.0-1 and the addition of DCD Tier 2, Section 2.0.2 “References” have no safety significance and that these changes remain within the acceptance criteria of the SRP Section 3.4.2, Revision 2.

#### *2.6.2.4 Conclusion*

The staff reviewed the changes to DCD Tier 2, Section 2.0 and Table 2.0-1 of the ABWR DCD that clarified the basis for the maximum groundwater and flood (or tsunami) level site parameters and determined that the changes conform to all applicable acceptance criteria as referenced in SRP Section 3.4.2 and to GDC 2, and 10 CFR § 52.47(a)(1)(iii).

### 2.6.8 Requirements for Determination of ABWR Site Acceptability

#### *2.6.8.1 Regulatory Criteria*

ABWR DCD Tier 2, Section 2.2, Revision 7, provides site parameters that are requirements for site acceptability that combined operating license (COL) applicants that reference the ABWR design must demonstrate are met. These site parameters cover both the evaluation of the radiological consequences of design-basis accidents (DBAs) for the siting and safety assessment, and the assessment of the radiological dose impacts of severe accidents. DCD Tier 2, Section 2.2.1, provides information related to DBAs, while ABWR DCD Tier 2, Section 2.2.2, gives information needed to perform severe accident consequence assessment. ABWR DCD Tier 2, Section 2.3.3, provides the related COL information items.

The change to the ABWR DCD does not alter the site parameters but modifies the ABWR DCD Tier 2, Section 2.2.2, discussion of how the COL applicant is to demonstrate that the severe accident site parameters are met. Specifically, instead of specifying use of the Calculation of Reactor Accident Consequences, Version 2, (CRAC 2) computer code, the revised text provides flexibility for the COL applicant to use a more modern severe accident consequence computer code. Since the applicant's design change is to provide DCD flexibility for a future COL applicant, it is an "amendment," as this term is defined in Chapter 1 of this supplement and is evaluated using the regulations in effect at renewal. The following regulatory requirements provide the basis for the acceptance criteria for the staff's review:

- 10 CFR § 52.47(a)(1), which requires site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters.

#### *2.6.8.2 Summary of Technical Information*

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. GEH proposed design changes to address Item No. 3 of the NRC July 20, 2012, letter, which suggested that the applicant consider removing references in DCD Tier 2, Chapter 2 directing COL applicants to use the CRAC 2 computer code,[5] which is no longer in use, and replace the references to the CRAC 2 computer code with generalized direction to use an appropriate severe accident consequences code such as the MELCOR Accident

[5] See NUREG/CR-2326, "Calculations of Reactor Accident Consequences Version 2, CRAC2: Computer Code, User's Guide," issued February 1983.

Consequence Code System (MACCS2).[6] In a letter dated June 19, 2015 (ADAMS Accession No. ML15170A039), GEH proposed changes to DCD Tier 2, Sections 2.2.2 and 2.3.3, to remove such references to the CRAC 2 severe accident consequences code and replace them with a generalized reference to severe accident consequence codes or more specifically to MACCS2 as an example. As noted in the revised text of paragraph three in DCD Tier 2, Section 2.2.2, when supplying the ABWR design data to be used in severe accident consequence assessment provided in DCD Tier 2, Table 2.2-2, and the tables in DCD Tier 2, Appendix 2A, the applicant retained the information in the CRAC 2 data input format as an example. GEH also made a conforming change to DCD Tier 2, Table 1.9.1, to revise the name of COL Information Item 2.42 to read "Severe Accident Consequence Computer Code Calculations." The applicant incorporated these changes in the ABWR DCD, Revision 6.

#### *2.6.8.3 Technical Evaluation*

The changes to DCD Tier 2, Table 1.9.1 and DCD Tier 2, Sections 2.2.2 and 2.3.3 remove certain references to a severe accident consequence computer code (CRAC 2) that is not currently in use by NRC staff or reactor licensees and applicants. The CRAC 2 code is an NRC-developed severe accident consequence computer code that has been used for environmental assessment and reactor safety studies. The MACCS code, developed in 1998 for reactor severe accident environmental assessments and reactor safety studies, is the only consequence code that the NRC staff uses for these assessments. MACCS is also used by power reactor licensees and applicants. Because 10 CFR § 52.47(a)(1) does not require that the DC specify the method that the COL applicant must use in determining site characteristics, the staff finds that the use in an ABWR COL application of an appropriate severe accident consequence computer code other than CRAC 2 is acceptable. In addition, if a COL applicant uses a code other than the MACCS2 code updated in 2004, as identified in the ABWR renewal DCD, the staff will assess the use of such other code against the review standards in effect at the time of the COL application, as appropriate. The changes to the ABWR DCD described above do not revise any accident analyses previously reviewed and found acceptable by the staff and do not affect any previous staff findings of reasonable assurance of adequate protection of public health and safety related to the ABWR design. The changes to the information regarding severe accident consequence assessment in the ABWR DCD, Revision 7, prevent the need for a COL applicant to justify a departure from the DCD information in order to use a state-of-the-art severe accident consequence code. Therefore, the staff finds acceptable the changes to DCD Tier 2, Table 1.9.1, COL Information Item 2.42, and Sections 2.2.2 and 2.2.3.

#### *2.6.8.4 Conclusion*

Based on the staff's review discussed above, the staff finds that the changes to ABWR DCD Tier 2 that provide adequate and sufficient information for COL applicants related to the use of severe accident consequence computer codes comply with 10 CFR § 52.47(a)(1) and are acceptable.

[6] MACCS2 is a fully integrated, engineering-level computer code developed at Sandia National Laboratories for the NRC. MACCS2 simulates the impact of severe accidents at nuclear power plants on the surrounding environment.

# 3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

### 3.2.3 Safety Classifications

In a letter dated August 25, 2015 (ADAMS Accession No. ML15237A192) the applicant proposed to add ABWR design enhancements to the spent fuel pool (SFP) level instruments that conform with applicable guidance specified in the Japan Lesson-Learned Project Directorate-Interim Staff Guidance (JLD-ISG)-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," which endorses with exceptions and clarifications the methodologies described in the Nuclear Energy Institute (NEI) industry guidance document NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation." This change to the design of SFP instruments resulted in a change to ABWR DCD Tier 2, Chapter 3, Table 3.2-1, that added SFP instrumentation to the table and identified the instrumentation's location, safety and quality classifications, and seismic categorization.

In addition, this change resulted in changes to the following ABWR DCD sections:

- Tier 1, Section 2.6.2, "Fuel Pool Cooling and Cleanup System," including Figure 2.6.2 and Table 2.6.2; and
- Tier 2, Chapter 1, Tables 1.8-21 and 1.8-22

These ABWR design enhancements would provide a potential COL applicant the means for meeting requirements of 10 CFR § 50.155, "Mitigation of Beyond-Design Basis Events," regarding requirements for safety-related SFP instrumentation which codified the requirements stemming from Commission Order EA-12-051.

Section 22.2 of this FSER supplement provides the staff's review of these changes and other changes associated with the new SFP instrumentation.

## 3.3 Wind and Tornado Loadings

### *3.3.1 Regulatory Criteria*

In accordance with NRC regulations, nuclear plants must be designed so that they remain in a safe condition under extreme meteorological events, including those that could result in the most extreme wind events (tornadoes and hurricanes) that could reasonably be predicted to occur. In the GEH ABWR DCD, Revision 7, the applicant added hurricane wind speed and hurricane missile spectra to the list of site parameter values presented in DCD Tier 1, Section 5.0, and DCD Tier 2, Section 2.0, of the GEH ABWR DCD. A combined license (COL) applicant that references the ABWR DC will assess whether the actual site characteristics fall within the site parameters specified for the ABWR design.

The applicant is making the changes to provide criteria for a COL applicant to determine whether an ABWR located at a particular site is appropriately protected against the effects of hurricane winds and missiles. In a letter dated September 25, 2014, the staff issued RAI 02-1, to the applicant, (ADAMS Accession No. ML14267A352), raising concerns about compliance with GDC 2 (1997) and 4 (1997) for hurricane loads and hurricane-generated missiles. In response, the applicant added information to address hurricane winds and missiles in DCD Tier 1 and Tier 2. Since the applicant's changes were in response to the staff's concerns regarding compliance with the regulations in effect at initial certification, these changes are "modifications," as described in Chapter 1 of this FSER supplement, and the staff will therefore evaluate them using the regulations applicable and in effect at the time of initial certification.

As a result of adding hurricane wind and missile site parameters, GEH updated the ABWR DCD to account for extreme hurricane wind and missile loading consistent with the methodology applicable at the time of initial certification. This evaluation documents the staff's review of these changes.

The relevant NRC requirements associated with the review of DCD Tier 2, Sections 3.3.1, "Severe Wind Loads," and DCD Tier 2, 3.3.2, "Extreme Wind Loads (Hurricanes and Tornados)," are given in GDC 2 (1997) and summarized below. The associated acceptance criteria are provided in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Sections 3.3.1 "Wind Loadings," and 3.3.2, "Tornado Loadings," Revision 2, 1981. The staff also considered the guidance in Regulatory Guide (RG) 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Revision 0, October 2011, which reflects an understanding of hurricane winds and missiles that was not reflected in earlier guidance.

- GDC 2 (1997), "Design Bases for Protection Against Natural Phenomena," requires, in part, that structures systems and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety function.

### *3.3.2 Summary of Technnical Information*

ABWR DCD, Revision 5, which GEH originally submitted in support of the ABWR DC renewal application, contained tornado site parameters related to the maximum tornado wind speed and missile spectra, but it did not contain any site parameters related to hurricane wind speed or hurricane missiles.

In Revision 6 to DCD Tier 2, Section 3.3.2.2, "Determination of Forces on Structures," the applicant proposed the inclusion of design-basis hurricane wind and missile loading.

- DCD Tier 1, Table 5.0, "ABWR Site Parameters," includes the addition of hurricane wind speed and missile spectra for the potential site.
- DCD Tier 2, Table 2.0-1, "Envelope of ABWR Standard Plant Site Design Parameters," includes the addition of hurricane wind speed and missile spectra for the potential site.

In Supplement 5, of its response, to RAI 02-1, dated April 13, 2017 (ADAMS Accession No. ML17103A125), following the submission of ABWR DCD, Revision 6, the applicant modified the DCD Tier 2, Table 2.0-1, to define the design-basis maximum hurricane wind speed of 257 kilometers per hour (km/h) as a fastest-mile wind speed which corresponds to 286.5 km/h 3-second gust wind speed in accordance with RG 1.221, Revision 0, measured at 10 meters above ground over open terrain. The applicant also clarified the maximum tornado wind speed of 483 km/h as a fastest quarter-mile (1/4-mile) wind speed which corresponds to 483 km/h 3-second gust wind speed.

### *3.3.3 Technical Evaluation*

The staff's evaluation of the missiles generated by extreme winds (hurricane) is provided in Section 3.5.1.4, "Missiles Generated by Natural Phenomena," of this FSER supplement and the staff's complete evaluation of meteorological site parameters is evaluated in Section 2.3 of this FSER supplement. In this FSER supplement section the staff evaluates the resulting hurricane wind and missile loading.

In the RAI dated September 25, 2014, the staff asked GEH to address the possibility that the wind speeds from the design-basis tornado may not be bounding for ABWR SSCs in certain locations along the United States Gulf Coast and the southern Atlantic Coast. In a letter dated November 19, 2014 (ADAMS Accession No. ML14324A084), GEH submitted its proposed changes to show that SSCs important to safety are protected from the effects of hurricane winds and missiles. In addition, GEH updated its RAI response in the following RAI supplements as follows:

- Supplement 1 by letter dated June 26, 2015 (ADAMS Accession No. ML15177A036)
- Supplement 2 by letter dated November 5, 2015 (ADAMS Accession No. ML15309A158)
- Supplement 3 by letter dated January 12, 2016 (ADAMS Accession No. ML16012A290)
- Supplement 4 by letter dated November 16, 2016 (ADAMS Accession No. ML16321A413)
- Supplement 5 by letter dated April 13, 2017 (ADAMS Accession No. ML17103A124)

In its responses to RAI 02-1, the applicant provided up-to-date hazards information in its ABWR DCD using current staff guidance with respect to hurricane wind speed and hurricane missiles based on RG 1.221, Revision 0.

The staff reviewed the changes to DCD Tier 1, Table 5.0 and Table 2.0-1, and DCD Tier 2 Sections 3.3.1 and 3.3.2, in order to determine compliance with GDC 2 (1997) using the guidance in SRP Sections 3.3.1, Revision 2 and Section 3.3.2, Revision 2.

The staff reviewed DCD Tier 1, Table 5.0 and DCD Tier 2, Table 2.0-1, and compared the design-basis hurricane wind speed and its missile velocities with the design-basis tornado wind speed and its missile velocities. The staff found that the design-basis hurricane wind speed and its missile velocities are bounded by the design-basis tornado wind speed and its missile velocities. The staff also reviewed the RG 1.221, Revision 0, and found that the methodology used in combining the effects of the design-basis hurricane winds and hurricane-generated missiles is the same as the one for the design-basis tornado winds and tornado-generated missiles in the original certification. Therefore, the staff concluded that the design-basis tornado loading governs as described in the original certification.

In addition, the staff reviewed DCD Tier 2, Section 3.3.1, and confirmed that the ABWR design-basis code, the American National Standards Institute (ANSI)/American Society of Civil Engineers (ASCE) 7-88, 1990, “Minimum Design Loads for Buildings and Other Structures,” issued October 5, 2018, was not changed. The staff also reviewed DCD Tier 2, Section 3.3.2.2, and found that the procedures for transforming the extreme hurricane wind loading into effective load distribution across the structures are consistent with those of the ABWR design-basis code, ANSI/ASCE, 7-88, 1990, which was approved in the original certification, and therefore are acceptable.

In a public teleconference on March 2, 2017, the staff requested further clarification on the ABWR DCD wind parameters in order to be consistent with the guidance for a design-basis hurricane wind speed in RG 1.221, Revision 0, based on the nominal 3-second peak-gust values at a height of 10 meters in flat open terrain, which is consistent with the definition of design wind speeds in the ANSI/ASCE, 7-88, 1990, design standard.

In GEH RAI Supplement 5, the applicant provided additional DCD changes to indicate the severe wind and extreme hurricane wind speed in terms of ”fastest-mile”, consistent with the ANSI/ASCE 7-88, 1990, methodology at the time of original certification. The corresponding equivalent ”3-second gust” is provided in the site-parameter table to facilitate comparison of design wind speeds consistent with RG 1.221, Revision 0.

Additionally, for tornado wind speed, the applicant updated the ABWR DCD to confirm the tornado design wind speed in the ”fastest 1/4-mile”. The corresponding equivalent ”3-second gust” design wind speed is also provided in the site-parameter table for future COL applicants’ site-specific tornado wind speeds.

The applicant provided the necessary wind speed information in the ABWR DCD, Revision 7, which incorporated the appropriate changes described in the applicant’s response to RAI 02-1, Supplement 5. Therefore, Confirmatory Item 3.3-1 from the staff advanced safety evaluation report with no open items for the ABWR DC renewal is resolved and closed.

### *3.3.4 Conclusion*

Based on the evaluation provided in this FSER section supplement, and as reviewed by the staff in accordance with the acceptance criteria in SRP Section 3.3.1, Revision 2, and Section 3.3.2, Revision 2, the staff concludes that the changes to the ABWR DCD, Revision 7, are acceptable and do not alter the safety findings made in NUREG–1503. The changes meet the applicable regulations in effect at initial certification including the requirements of GDC 2 (1997).

### *3.5.1.4 Missiles Generated by Natural Phenomena*

#### *3.5.1.4.1 Regulatory Criteria*

In this FSER supplemental section the staff reviews the ABWR DCD, Revision 7, and evaluates the applicant’s assessment of possible hazards attributable to missiles generated by hurricanes to ensure that the applicant has chosen and properly characterized appropriate design-basis missiles. The applicant provided additional information to address hurricane-generated missiles for the GEH ABWR DC renewal to clarify the possible hazards attributable to missiles generated by hurricanes. In a letter dated September 25, 2014, the staff issued RAI 02-1, to the applicant, (ADAMS Accession No. ML14267A352), raising concerns about compliance with GDC 2 (1997) and 4 (1997) for hurricane loads and hurricane-generated missiles. In response, the applicant added information to DCD Tier 1, Section 5.0 and Tier 2, Section 2.0. Since the applicant's changes were in response to the staff's concerns regarding compliance with regulations in effect at initial certification, these changes are "modifications," as described in Chapter 1 of this staff FSER supplement and will correspondingly be evaluated using the regulations applicable and in effect at the time of the initial ABWR certification.

The relevant NRC requirements associated with the review of the changes are summarized below. The associated acceptance criteria are given in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 3.5.1.4, "Missiles Generated by Natural Phenomena" Revision 2, 1981. Review interfaces with other SRP sections can also be found in SRP Section 3.5.1.4.I (1981).

- GDC 2 (1997) "Design Bases for Protection Against Natural Phenomena," requires, in part, that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety function.
- GDC 4 (1997) "Environmental and Dynamic Effects Design Bases," requires, in part, that SSCs important to safety shall be appropriately protected against dynamic effects, including the effects of missiles that may result from equipment failures and from events and conditions outside the nuclear power unit.

*3.5.1.4.2 Summary of Techncial Information*

ABWR DCD, Revision 5, which was originally submitted in support of the ABWR DC renewal application, contained tornado site parameters related to the maximum tornado wind speed and missile spectra, but did not contain any site parameters related to hurricane wind speed or hurricane missiles.

The applicant included the following changes for the ABWR DC Renewal application in ABWR DCD Revision 6:

- DCD Tier 1, Table 5.0, "ABWR Site Parameters," included changes to address hurricane missiles.
- DCD Tier 2, Table 2.0-1, "Envelope of ABWR Standard Plant Site Design Parameters," and Section 3.5.1.4, "Missiles Generated by Natural Phenomena," included changes that describe the spectrum of missiles generated by hurricane winds and their associated velocities.

DCD Tier 2, Table 2.0-1, describes the design-basis hurricane missile spectra for the GEH ABWR design as follows:

> a rigid missile that tests penetration resistance, such as a 130 kg (287 lb), 20 cm (7.9 in.) diameter armor piercing shell
> a small rigid missile of a size that is sufficient to pass through openings in protective barriers, such as a 2.54 cm (1 in.) diameter solid steel sphere

These missiles all have a horizontal hurricane missile velocity of 59 percent of the maximum hurricane wind speed. In addition, the ABWR DCD, Revision 6, markup to DCD Tier 2, Table 2.0-1 (submitted by GEH as discussed in the technical evaluation section of this SER) states that all missiles have a vertical hurricane missile velocity of 26 meters per second (m/s) (58 miles per hour (mph)).

The applicant assumed that the automobile missile impacts at all altitudes is less than 9.14 meters (m) (30 feet (ft.)) above plant grade within 0.8 kilometer (km) (0.5 mile (mi)) of the plant structures, in accordance with the guidance of SRP Section 3.5.1.4 and Regulatory Guide (RG) 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” Revision 0, issued October 2011. In addition, the applicant included a combined license (COL) Information Item in DCD Tier 2, Section 3.5.3, for the COL applicant to confirm there are no elevated parking lots within 0.8 km (0.5 mi) of the plant structures that can cause an automobile impact higher than 9.14 m (30 ft.) above plant grade.

Footnotes added by the applicant as part of ABWR DCD, Revision 7, to both DCD Tier 1, Table 5.0 and DCD Tier 2, Table 2.0-1 state that 257 km/h is a fastest-mile wind speed which corresponds to 286.5 km/h 3-second gust wind speed, as the design-basis hurricane wind speed parameter for the ABWR DC, in accordance with RG 1.221, Revision 0, measured at 10 m above ground over open terrain. The staff notes that a wind speed of 286.5 km/h is equivalent to 178 mph or 79.6 m/s.

*3.5.1.4.3 Technical Evaluation*

In this supplemental FSER section, the staff evaluates the hurricane missile parameters for the ABWR DC renewal. Supplemental FSER Sections 2.3.1 and 3.3 provide the staff’s evaluation of the hurricane winds and the resulting extreme wind loadings on structures important to safety, respectively.

In the RAI dated September 25, 2014, the staff requested that GEH update its ABWR DCD during the renewal process to address the possibility that the wind speeds from the design-basis tornado may not be bounding for ABWR SSCs in certain locations along the United States Gulf Coast and the southern Atlantic Coast. The study of missile speeds during hurricanes, NUREG/CR-7005, “Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants,” issued November 2011 (ADAMS Accession No. ML11335A031), concluded that, because of assumed differences between the tornado and hurricane wind fields, airborne missiles can fly faster in a hurricane wind field with the same 3-second gust wind speed at 10 m (33 ft) above ground as a tornado wind field. Missiles in a hurricane wind field may have higher maximum velocities than in a tornado wind field because hurricane missiles are subject to high wind speeds throughout their trajectory.

In response, the applicant provided updated hazards information in its ABWR DCD with respect to hurricane missiles based on RG 1.221, Revision 0.

In its RAI response letter dated, November 19, 2014 (ADAMS Accession No. ML14324A084), GEH submitted its changes to show that SSCs important to safety are protected from the effects of hurricane winds and missiles. In addition, GEH updated its RAI response in the following RAI supplements as follows:

- Supplement 1 by letter dated June 26, 2015 (ADAMS Accession No. ML15177A036)
- Supplement 2 by letter dated November 5, 2015 (ADAMS Accession No. ML15309A158)
- Supplement 3 by letter dated January 12, 2016 (ADAMS Accession No. ML16012A290)
- Supplement 4 by letter dated November 16, 2016 (ADAMS Accession No. ML16321A413)

The applicant's RAI supplements were based on feedback from staff at public meetings held with GEH on their initial response to RAI 02-1, dated November 19, 2014. These public meetings took place on May 7, 2015 (ADAMS Accession No. ML15162A613), October 15, 2015 (ADAMS Accession No. ML15306A104) and October 27, 2016 (ADAMS Accession No. ML17004A316).

In its supplemental responses to RAI 02-1, the applicant provided additional changes to the ABWR DCD to address hurricane winds and associated missiles as an update and modification to the ABWR DCD, Revision 6.

The staff reviewed the additional changes as presented in RAI 02-1, Supplement 4, ABWR DCD, Revision 6 markups related to the design bases for the missile spectra.

Protection from a spectrum of missiles with the critical characteristics set forth in RG 1.221, Revision 0, provides assurance that the necessary SSCs will be available to mitigate the potential effects of hurricane winds and missiles on plant SSCs important to safety. RG 1.221, Revision 0, provides contour maps of U.S. coastal areas most susceptible to hurricanes and associated design-basis wind and missile speeds. The staff reviewed the information submitted by the applicant and finds the hurricane generated missile spectra and hurricane missile velocities to be either consistent or conservative with respect to the guidance of RG 1.221, Revision 0. In addition, the design-basis hurricane missile velocities presented in the revised ABWR DCD, Revision 7, are bounded by the tornado missile velocities already included in the original ABWR DC.

Based on its review, the staff finds that the applicant's changes meet the guidance in RG 1.221, Revision 0, for design-basis hurricane missiles. Therefore, the staff concludes that the ABWR hurricane missile parameters meet the requirements of GDC 2 and GDC 4 in effect at initial certification with respect to hurricane generated missiles.

The applicant provided the necessary hurricane parameters and hurricane-generated missile spectra in the ABWR DCD, Revision 7, which incorporates the appropriate changes described in the applicant's response to RAI 02-1, Supplement 4. Therefore, Confirmatory Item 3.5.1-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*3.5.1.4.4 Conclusion*

As discussed above, the staff's review concludes that the applicant's changes to its design-basis hurricane parameters and hurricane-generated missile spectra for the GEH ABWR design meet the guidance in RG 1.221, Revision 0, for design-basis hurricane wind borne missiles for nuclear power plants, and therefore are acceptable. Based on the evaluation provided in this supplemental FSER section to NUREG–1503, the staff concludes that the changes to the ABWR DCD, Revision 7, are acceptable, do not alter the safety findings made in NUREG–1503 for the ABWR DC and meet the applicable regulations in effect at initial certification, including the requirements of GDC 2 (1997) and GDC 4 (1997), as reviewed by the staff in accordance with the associated acceptance criteria in SRP Section 3.5.1.4, Revision 2.

### 3.7.3 Seismic Subsystem Analysis

#### *3.7.3.1 Regulatory Criteria*

The certified ABWR DCD and the ABWR DCD, Revision 5, for the renewal application, did not provide information regarding the design and analysis of the tunnel structures for diesel generator fuel oil transfer systems (DGFOTS). In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. In Item No. 23 of the July 20, 2012, letter, the staff requested that GEH provide the tunnel structure analysis. The applicant addressed this omission by providing proposed changes to the ABWR DCD in a letter dated July 17, 2015 (ADAMS Accession No. ML15198A344), to explicitly specify the Seismic Category I tunnel structures to be a reinforced concrete tunnel and be designed and analyzed according to NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 3.7.3, "Seismic Subsystem Analysis," Revision 2, issued August 1989. In addition, GEH proposed to add a combined license (COL) information item directing COL applicants referencing the ABWR design to provide the design and analysis for the DGFOTS tunnel structures for routing the fuel oil transfer piping and cable system from the fuel oil storage tank to the diesel generator (DG) in the reactor building.

As a result, the applicant stated that it is not changing the ABWR design, but rather specifying the classification and criteria consistent with SRP Section 3.7.3, Revision 2, to be used by COL applicants to design and analyze the tunnel structures. The proposed revisions effectively constitute interface requirements that should have been identified in the initial DCD to satisfy 10 CFR § 52.47(a)(1)(vii) (1997 version). GEH also proposed clarifications to ABWR DCD Tier 2, Sections 3.8.4.1.3 and 3.12.3 that are unrelated to the DGFOTS tunnel structures. These clarifications are consistent with the original understanding of the design information in the initial DC. Therefore, the proposed changes are "modifications," in accordance with 10 CFR § 52.59(a) as this term is defined in Chapter 1 of this supplement, therefore the change complies with regulations applicable and in effect at the time the certification was issued.

The following regulatory requirements apply to the evaluation of the proposed GEH ABWR DCD modifications:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 2, "Design Bases for Protection Against Natural Phenomena," (1997) in the relevant parts, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as earthquakes, without loss of capacity to perform their intended safety functions. GDC 2 further requires that the design bases reflect appropriate consideration of the most severe natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated in the past.

- 10 CFR Part 100, “Reactor Site Criteria,” Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” (1997) requires considering two earthquake levels, the safe shutdown earthquake (SSE) and operating basis earthquake, in the design of safety-related SSCs. Appendix A to 10 CFR Part 100 further states that the engineering method used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the SSE, shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that SSCs can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.
- 10 CFR § 52.47, “Content of Applications,” (1997) the NRC states in 10 CFR § 52.47(a)(1)(vii) that the interface requirements must be met by those portions of the plant for which the application does not seek certification. Also, 10 CFR § 52.47(a)(1)(viii) requires justification that compliance with the interface requirements of this section is verifiable through inspections, tests, or analyses, and requires the method to be used for verification of interface requirements to be included as part of the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) required by 10 CFR § 52.47(a)(1)(vi).

### *3.7.3.2 Summary of Technical Information*

GEH submitted the proposed changes in the July 17, 2015 letter, to indicate that the tunnel structures for DGFOTS are Seismic Category I. In addition, GEH proposed a COL Information Item directing COL applicants to provide the design and analysis of the tunnel structures in accordance with SRP Section 3.7.3, Revision 2. GEH also proposed clarifications to ABWR DCD Tier 2, Sections 3.8.4.1.3 and 3.12.3, that are unrelated to the DGFOTS tunnel structures.

### *3.7.3.3 Technical Evaluation*

The staff reviewed the proposed changes to ABWR DCD Tier 2, Revision 6, Table 1.9-1 and Sections 3.7.3.12, 3.8.4.1.3, 3.12.2.1, and 3.12.3, as well as new Sections 3.8.4.1.6 and 3.8.6.5, in order to determine compliance with GDC 2 and 10 CFR Part 100, Appendix A. The staff used the review guidance in SRP Section 3.7.3, Revision 2, to conduct its review.

In the July 17, 2015 letter, GEH added the COL information item for Seismic Category I buried piping, conduits and tunnels to the list of COL information items in ABWR DCD Tier 2, Table 1.9-1. In DCD Tier 2, Section 3.7.3.12, GEH added text to describe the physical characteristics and design/analysis specifications of Seismic Category I underground tunnels as reinforced concrete structures in direct contact with soil and having adequate dynamic clearance to their housing piping/cables to avoid transmission of seismic in-ground accelerations and displacements. GEH also added the statement that the design and analysis of Seismic Category I underground tunnels follow the engineering process specified in SRP Section 3.7.3, Revision 2. In the new DCD Tier 2, Section 3.8.4.1.6, GEH added that the Seismic Category I buried piping, conduits and tunnels, shall be designed and analyzed in accordance with SRP Section 3.7.3, Revision 2. In the new DCD Tier 2, Section 3.8.6.5, GEH stated that the COL applicant shall provide a design and analysis report for Seismic Category I buried piping, conduits and tunnels in accordance with SRP Section 3.7.3, Revision 2, and referred to DCD Tier 2, Section 3.7.3.12. GEH also described in DCD Tier 2, Section 3.12.2.1, that specific seismic requirements are included in Section 3.7.3.12 and specified in SRP Section 3.7.3.

The staff reviewed the proposed changes in DCD Tier 2, Table 1.9-1, Sections 3.7.3.12, 3.8.4.1.6, and 3.8.6.5, and concludes that it is not practical to perform Seismic Category I assessment for the tunnel structures at this stage because site-specific soil information is not available. Therefore, the staff finds it acceptable to defer the design and analysis of Seismic Category I tunnel structures to the COL applicant because (1) the proposed changes provide assurance that the Seismic Category I tunnel structures will not be adversely affected by the adjacent Diesel Oil Storage Tank and Reactor Building structures under design-basis loads, and (2) the design and analysis of the tunnel structures will be performed by the COL applicant per the guidelines provided in SRP Section 3.7.3, Revision 2.

In its letter dated July 17, 2015 (ADAMS Accession No. ML15198A344), GEH also added the text “rebar stress and required rebar” to ABWR DCD Tier 2, Section 3.8.4.1.3, that included rebar information for the radwaste building (RW/B). In addition, GEH added the text “Non-Safety Related” to the title of DCD Tier 2, Section 3.12.3. The staff reviewed the these editorial changes in DCD Tier 2, Sections 3.8.4.1.3 and 3.12.3, and finds that they are not relevant to the Seismic Category I tunnel structures of DGFOTS; however, they are acceptable to the staff because: (1) GEH follows the common engineering principles and practices, which clarify the rebar information for the RW/B; and (2) the change of the title of DCD Tier 2, Section 3.12.3 is consistent with the contents of this section.

### *3.7.3.4 Conclusion*

Based on the evaluation provided in this FSER supplement, the staff concludes that the COL Information Item directing COL applicants to provide the design and analysis of the tunnel structures for DGFOTS, will assure that all Seismic Category I utilities (i.e., piping, conduits, or auxiliary system components) that are routed within these tunnels are adequately protected and will perform their intended safety functions. Therefore, the staff concludes that the requirements of GDC 2, 10 CFR Part 100, Appendix A and 10 CFR § 52.47 are met.

# 4 REACTOR

## 4.2 Fuel System Design

### *4.2.1 Regulatory Criteria*

In the ABWR DCD, Revision 7, GEH proposed to include additional clarity in the ABWR DCD concerning a combined license (COL) applicant's responsibility to perform an analysis of the combined loading on the reactor core from a seismic event and loss-of-coolant-accident (LOCA) to demonstrate conformance to the structural acceptance requirements for the reactor core.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 18a of the letter, the staff requested the applicant to provide the analysis of the combined seismic and LOCA loading on the reactor core to demonstrate conformance to the structural acceptance criteria described in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition (SRP) Section 4.2, "Fuel System Design," Appendix A, "Evaluation of Fuel Assembly Structural Responses to Externally Applied Forces."

In a letter dated September 24, 2015 (ADAMS Accession No. ML15271A169), the applicant stated that the DCD need not be revised because the criteria of SRP Section 4.2, Appendix A, are directly satisfied by a requirement in ABWR DCD Tier 2, Chapter 4, Section 4.2.3.1.2(1). However, to provide further clarity for potential COL applicants in the future, GEH added COL Information Item 4.2.5.2 to DCD Tier 2, Revision 6, Section 4.2.5, "COL License Information."

Because the applicant's proposed change clarifies information in the original ABWR design certification, it is a "modification," as this term is defined in Chapter 1 of this supplement. Therefore, this modification must comply with the Atomic Energy Act of 1954, as amended, and the Commission's regulations applicable and in effect at the time the certification was originally issued. Therefore, the staff evaluated the proposed change using the regulations in effect at the time the certification was originally issued.

Appendix A to SRP Section 4.2, Revision 2, issued July 1981, describes the relevant requirements for this area of review and the associated acceptance criteria. These requirements appear in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the structural protection for fuel assemblies and control blades during accidents involving earthquakes. GDC 2 requires the design bases of structures, systems, and components, which include fuel assemblies and control blades, to reflect appropriate consideration of natural phenomena, including consideration of combined loading due to natural phenomena and limiting hydrodynamic loads.

### *4.2.2 Summary of Technical Information*

In its letter dated September 24, 2015, GEH proposed a resolution to Item No. 18a of the staff letter dated July 20, 2012. GEH submitted the proposed ABWR DCD, Revision 6, markups in Enclosure 2 of the September 24, 2015, letter to address the staff's request.

In Enclosure 1 of that letter, GEH described the proposed changes it would make to the ABWR DCD Tier 2, Subsection 4.2.5 and Table 1.9-1 to include the new COL Information Item clarifying the responsibility of future COL applicants regarding analysis of the combined seismic and LOCA loading on the reactor fuel.

### *4.2.3 Technical Evaluation*

The ABWR DCD reference fuel is GE P8x8R as described in GE Topical Report NEDE-31152P, "General Electric Fuel Bundle Designs Evaluated with GESTAR-Mechanical Analysis Bases (proprietary)," dated December 1988, which used the fuel bundle design methodologies described in GE Topical Report NEDE-24011-P (GESTAR II), Amendment 7. The staff approved the fuel design methodologies in GESTAR II, Amendment 7, in an NRC safety evaluation letter dated March 1, 1985, from C. O. Thomas to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P Amendment 7 to Revision 6, General Electric Standard Application for Reactor Fuel" (ADAMS Accession No. ML090760583 (non-public)).

ABWR DCD Tier 2, Section 4.2.3.1.1, describes these approved, referenced fuel design methodologies. Additionally, DCD Tier 2, Appendix 4B lists the fuel licensing acceptance criteria and Appendix 4D demonstrates that the reference fuel meets the acceptance criteria. The GESTAR II, Amendment 7, references the seismic-and-LOCA loading evaluation in GE Topical Report NEDE-21175-3-P, “BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings,” Amendment 3, issued July 1982 (proprietary). In the NRC safety evaluation for GESTAR II, Amendment 7, the NRC stated the following:

> *The entire seismic-and-LOCA loads evaluation (including design limits) has been described by GE in the approved topical report NEDE-21175-3 to which GESTAR II makes reference. We conclude that the criteria for fuel assembly structural damage from external forces in NEDE-21175-3 are acceptable for GESTAR II.*

In ABWR DCD Tier 2, Section 4.2, GEH stated that each COL applicant may have different fuel and core designs that the COL applicant will provide to the NRC for review and approval. In Section 4.2 of NUREG–1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, issued July 1994 (the final safety evaluation report (FSER) for the original certification of the ABWR design), the NRC approved the ABWR fuel design with the following condition in DCD Tier 2, Section 4.2.3.1.2:

> *The license/applicant must provide a plant-specific analysis of combined seismic and LOCA loading using NRC-approved methodology or another acceptable method to demonstrate conformance to the structural acceptance requirements described in Appendix A of Standard Review Plan Section 4.2.*

The staff notes that because this seismic and LOCA analysis is site-specific, deferring to the COL applicant to perform this analysis is also acceptable under current guidance in accordance with Regulatory Guide 1.206, “Applications for Nuclear Power Plants,” Revision 1, issued October 2018, Section C.1.11.b, “Supplemental Information.” In the September 24, 2015 letter responding to Item 18a, GEH proposed the following COL Information Item in DCD Tier 2, Section 4.2.5, “COL License Information” to ensure clarity concerning the COL applicant's responsibility to perform an analysis of reactor core combined seismic and LOCA loading:

> *4.2.5.1 - Reactor Core Seismic and LOCA Structural Acceptance*
> *The COL applicant shall provide the NRC a confirmatory plant-specific analysis of the reactor core combined seismic and LOCA loading using NRC-approved methodology or another acceptable method to demonstrate conformance to the structural acceptance requirements described in Appendix A of Standard Review Plan, Section 4.2, for the fuel referenced in the COL application. This analysis will use as input the site-specific ground motion and the fuel characteristics of the plant's initial core load.*

The staff evaluated the above information and determined that the NRC previously approved ABWR reference fuel and design methodologies, and that the proposed COL Information Item will add clarity to the ABWR DCD concerning the COL applicant's responsibility to perform an analysis of the combined seismic and LOCA loading on the reactor core that will meet the structural acceptance criteria in Appendix A to SRP Section 4.2. Therefore, GEH's response to Item 18a of the staff's letter dated July 20, 2012, is acceptable. In addition, the staff confirmed incorporation of the COL Information Item into DCD Tier 2, Revision 7 of Section 4.2.5.2.

### *4.2.4 Conclusion*

The staff reviewed the applicant's proposed changes to the ABWR DCD, Revision 7, as described above. Based on this evaluation, the staff concludes that the changes are acceptable because the ABWR reference fuel and methodologies continue to meet all applicable regulatory requirements at the time of original certification, including GDC 2 as referenced in Appendix A to SRP Section 4.2, Revision 2, July 1981, and the changes do not alter the safety conclusions made previously in the staff FSER as documented in NUREG–1503.

# 5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

## 5.2.5 Reactor Coolant Pressure Boundary Detection

### *5.2.5.1 Regulatory Criteria*

In this section the staff reviews and evaluates the applicant's proposed change to a combined license (COL) Information Item on reactor coolant pressure boundary (RCPB) leakage detection.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. In Item No. 12, the staff asked the applicant to revise a COL Information Item to develop operating procedures to respond to prolonged low-level-reactor coolant leakage below technical specification limits. GEH proposed to revise an existing COL Information Item in the ABWR DCD to provide additional details on the procedures associated with low-level-reactor coolant leakage to be developed by COL applicants.

This change would require a COL applicant to address the issue subject to the requirements as they exist at the time the COL application is filed. Therefore, in accordance with 10 CFR § 52.59(c), this design change is an "amendment," as this term is defined in Chapter 1 of this supplement and the staff will evaluate the proposed change using the regulations in effect at renewal.

The relevant requirements of the for this area of review, and the associated acceptance criteria, are given NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," (SRP) Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection," Revision 2, issued in March 2007, and are summarized below:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to the components which are part of the RCPB being designed, fabricated, erected, and tested to the highest quality standards practical. GDC 30 requires that means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage; and
- Regulatory Guide (RG) 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," issued May 2008, as it relates to the selection of RCPB leakage detection systems

In a letter dated May 27, 2015 (ADAMS Accession No. ML15147A593), GEH proposed to revise an existing COL Information Item as described below.

### *5.2.5.2 Summary of Technical Information*

In the certified ABWR DCD, Revision 5, COL Information Item 5.2.6.1 had stated:

> Procedures and graphs will be provided by the COL applicant to operations for converting the various indicators into a common leakage equivalent (DCD Section 5.2.5.9).

GEH proposed to revise COL Information Item 5.2.6.1, “Leak Detection Monitoring,” as part of their application to renew the ABWR DCD to be consistent with updated staff guidance as follows:

> The COL Applicant will include 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.
> - A milestone for completing this category of operating procedures.

Based on the proposed COL Information Item above, COL applicants referencing the renewed ABWR DCD, Revision 7, will be responsible for developing a procedure to convert different parameter indications for identified and unidentified leakage, including common leak rate equivalents (volumetric or mass flow) and leak rate rate-of-change values. Typical monitoring would include parameters such as sump pump run time, sump level, condensate transfer rate, and process chemistry / radioactivity. The monitored leakage equivalent would also provide information used by the plant operators to manage the leakage and establish whether the leakage rates are within the allowable technical specifications and determine the trend (DCD Tier 2, Section 5.2.5.9).

The staff additionally confirmed that the proposed change will result in COL applicants being responsible for the development of procedures for monitoring, recording, trending, determining the source(s) of leakage, and evaluating potential corrective action plans in accordance with the latest staff guidance. An unidentified leakage rate-of-change alarm would provide operators an early alert to initiate response actions before reaching the limiting technical specifications requirements.

GEH updated the COL Information Item listing in DCD Tier 2, Table 1.9-1 to reflect the changes described above; and the NRC staff confirmed that the applicant implemented the above changes in DCD Tier 2, Table 1.9-1 and DCD Tier 2, Section 5.2.6.1, in the ABWR DCD Revision 6 and as reflected in the ABWR DCD Revision 7.

### *5.2.5.3 Technical Evaluation*

Insights from operating experience indicate that prolonged low-level unidentified reactor coolant leakage inside containment could cause corrosion and material degradation such that it could compromise the integrity of a system leading to the gross rupture of the RCPB. In RG 1.45, Revision 1, the Regulatory Position on “Operations-Related Positions,” provides guidance to address the issue and meet the requirements of GDC 30. A COL applicant should establish procedures for responding to prolonged low-level-reactor coolant system (RCS) leakage. The procedures should specify operator actions in response to prolonged low-level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the technical specification (TS) limits in order to provide operators sufficient time to take action before the TS limit is reached. These procedures would include identifying, monitoring, trending, and managing prolonged low-level leakage.

In DCD Tier 2, COL Information Item 5.2.6.1, GEH revised the leak detection monitoring for a COL applicant, such that procedures, in regard to monitoring, recording, trending, determining the source(s) of leakage, and evaluating potential corrective action plans are developed, to guide the operator's response to RCPB leakage.

Based on the above, the staff determined that the applicant's proposed approach is consistent with the guidance in RG 1.45, Revision 1, pertaining to management of prolonged low-level RCS leakage. Therefore, the staff finds that the applicant's approach is acceptable.

*5.2.5.4 Conclusion*

Based on the evaluation provided in this SER section supplement, the staff concludes that the proposed amendment to the ABWR DCD associated with the revised COL Information Item meets the applicable guidance in RG 1.45, Revision 1, and therefore the requirements of GDC 30 as reviewed by the staff in accordance with SRP acceptance criteria in Section 5.2.5, Revision 2, of NUREG–0800 and therefore is acceptable.

## 5.4.7 Residual Heat Removal System

*5.4.7.1 Regulatory Criteria*

In the GEH ABWR DCD, Revision 7, the applicant made a change to add a redundant alternating current (ac) independent water addition (ACIWA) mode to the residual heat removal (RHR) system Loop B. The modification would provide emergency water injection from the fire protection system (FPS) or from an external water source such as a fire truck through a cross connection in the RHR Loop B to the reactor vessel, the containment wetwell or drywell spray sparger, or the spent fuel pool. The proposed additional ACIWA RHR Loop B is configured similarly to the current ABWR ACIWA RHR Loop C components and piping arrangement with equivalent system design parameters.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. In Item No. 26 of the letter, the applicant was requested to address ABWR DCD design changes related to aspects of the NRC Fukushima Near-Term Task Force Recommendation 4.2, regarding mitigation strategies for beyond-design-basis external events which was based on the NRC policy at that time. The policy on mitigation strategies, at that time, was outlined in a staff requirements memo SECY-12-0025, “Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami,” dated February 17, 2012, (ADAMS Accession No. ML12039A111).

In a public teleconference on March 17, 2016 (ADAMS Accession No. ML16124A049), the NRC staff requested that GEH clarify the ABWR response to a beyond-design-basis event with specific information items to be provided by a combined license (COL) applicant that would also address the draft mitigation of beyond-design-basis events (MBDBE) rule (10 CFR § 50.155, “Mitigation of beyond-design-basis events”), that was pending at that time. Subsequently, during the MBDBE rulemaking that created 10 CFR § 50.155, the Commission decided not to impose mitigation strategies requirements on

DCs.[7] The final rule was published in the *Federal Register* on August 9, 2019 (84 FR 39684) and became effective September 9, 2019.

In a letter dated January 23, 2017 (ADAMS Accession No. ML17025A386), GEH provided supplemental information for GEH's response to Item No. 26 of the July 20, 2012 staff letter. The applicant narrowed the scope of Item No. 26 to exclude changes directly related to 10 CFR § 50.155. As such, GEH retained the addition of the ACIWA RHR Loop B as an operational enhancement to provide additional defense-in-depth. These proposed ABWR design enhancements could provide a potential COL applicant the means for meeting MBDBE rule requirements.

These proposed changes do not fall within the definition of a "modification," as described in Chapter 1 of this FSER supplement. Therefore, in accordance with 10 CFR § 52.59(c), these design changes are "amendments," as this term is defined in Chapter 1 of this FSER supplement and the staff will evaluate the proposed changes using the regulations in effect at renewal. The regulatory requirements for evaluating the proposed DCD design changes to add an ACIWA subsystem to RHR Loop B and related changes are as follows:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 34, "Residual Heat Removal," as it relates to the ABWR RHR system, which requires the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are not exceeded. Compliance with GDC 34 enhances plant safety by providing assurance that decay and residual heat removal will be accomplished, and the reactor coolant system pressure boundary and fuel cladding integrity will be maintained, thereby minimizing the potential for the release of fission products to the environment.

The staff reviewed this amendment for renewal of the ABWR DC in accordance with NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 5.4.7, "Residual Heat Removal (RHR) System," Revision 5, issued May 2010. This ABWR DC renewal design change does not alter the previous staff safety findings regarding the ABWR RHR from NUREG–1503, the staff FSER for the ABWR DC, Chapter 5, Section 5.4.7.

#### *5.4.7.2 Summary of Technical Information*

In Section 5.4.7 of the ABWR FSER, NUREG–1503, the staff provided its technical evaluation and regulatory approval of the original ABWR ACIWA subsystem which has the function of providing a beyond- design-basis emergency water source to the reactor vessel, containment and spent fuel pool through the ABWR RHR Loop C from the plant FPS supplemental water sources. The staff described the ACIWA subsystem piping and components arrangement that links the FPS water source or an alternate external water source such as a fire truck to the RHR Loop C pump discharge line downstream of the pump's discharge check valve. The ACIWA safety-related isolation valves are normally closed and designed to isolate the non-safety FPS from the safety related RHR system. During a beyond-design-basis event including the loss of onsite and offsite ac (e.g.,

[7] In the MBDBE proposed rule regulatory analysis (ADAMS Accession No. ML15266A133), the Commission proposed to not make the MBDBE proposed rule applicable to existing DCs, which included the ABWR, because "[t]he issues that may be resolved in a DC and accorded issue finality may not include operational matters, such as the elements of the [MBDBE] proposed rule."

extended station blackout (SBO)), the valves can be operated manually and be placed into operation locally from the emergency core cooling systems (ECCS)/ RHR valve room. This flow path allows an additional water source for injection into the reactor vessel and the drywell during postulated beyond-design-basis conditions including an SBO condition where all ac power and all ECCS pumps are unavailable.

The GEH proposal to add a redundant ACIWA subsystem to the RHR Loop B discharge line, as described in the following Section 5.4.7.3 of this supplemental FSER, includes revisions to both DCD Tier 1, and DCD Tier 2, information.

### *5.4.7.3 Technical Evaluation*

In the ABWR DCD, Revision 7, includes an additional ACIWA subsystem to RHR Loop B as an enhancement to the design features that provide water makeup from the FPS or backup external source to the reactor pressure vessel, containment, and spent fuel pool during degraded beyond-design-basis plant conditions (such as an extended SBO) when both onsite and offsite ac power sources are unavailable. The staff reviewed the proposed change to RHR Loop B and considers the additional ACIWA comparable to the RHR Loop C ACIWA. Each ACIWA subsystem will have connections to the FPS adjacent to the ECCS/ RHR valve room with a check valve upstream of two normally locked closed safety related manually operated valves in series to isolate and prevent back flow into the FPS. The external source connection for the additional ACIWA subsystem to RHR Loop B is configured the same as the original ACIWA connection, with the exception of an additional manual valve located outside of the reactor building. The staff finds the configuration acceptable because it provides isolation from the safety related RHR system during normal operation while preventing reverse flow if the manual valves in series are mis-aligned during operation of the RHR. The addition of a redundant ACIWA mode to the RHR system does not alter the function of the RHR system and adds additional capability, therefore the original staff findings in NUREG–1503 are not altered and the requirements of GDC 34 are maintained and enhanced.

The staff also finds that the ACIWA vessel injection, containment injection, or spent fuel pool makeup modes are not adversely affected by this additional design enhancement and adds additional flexibility to the ACIWA system. The staff considers the physical separation of the two ACIWA subsystems sufficient to ensure that at least one will be available during degraded plant conditions such as an extended SBO. In addition, the staff finds the ABWR DCD, Revision 7, complete and confirmed the changes against the GEH markups from it’s January 23, 2017, letter, of the DCD Tier-1 and Tier-2 figures, sections, and tables.

### *5.4.7.4 Conclusion*

Based on the evaluation provided in this FSER section supplement, the staff concludes that the proposed ABWR DCD design enhancements do not alter the safety findings made in the FSER for the original ABWR certification. In addition, the changes proposed by the applicant improve the reliability of the ACIWA to deliver water makeup to the reactor vessel, containment, and spent fuel pool during degraded plant conditions. Therefore, the staff finds that the changes are in compliance with GDC 34 and the changes are acceptable because they improve plant operational flexibility and safety by providing additional means of residual and decay heat removal.

#### *5.4.7.1.1.10 ACIWA Mode*

##### *5.4.7.1.1.10.1 Regulatory Criteria*

In the ABWR DCD, Revision 7, GEH proposed a change to add an alternating current (ac) independent water addition (ACIWA) subsystem to Loop B of the ABWR residual heat removal (RHR) system, and to add the component designation "C" for the existing ACIWA subsystem components in Loop C of the RHR system. The ACIWA subsystem on Loops B and C of the RHR system consists of piping and valves that connect the non-safety/safe-shutdown portion of the fire protection system (FPS) to the safety-related RHR system to allow for injection of water into the reactor vessel, the drywell or wetwell spray header, or the spent fuel pool (SFP) during events when ac power is unavailable from both onsite and offsite sources. The safety-related portion of the ACIWA subsystem includes gate valves RHR-F101B/C and RHR-F102B/C (which isolate the FPS from the RHR system and are normally locked closed), instrument valves RHR-F790B/C, test connection valves RHR-F591B/C, and vent and drain valves RHR-F592B/C.

GEH also provided in DCD Tier 1, Section 2.4.4, "Reactor Core Isolation Cooling System (RCIC)," and Tier 2, Section 5.4.6.1.1.1, "Residual Heat" a design enhancement to the reactor core islolation cooling (RCIC) system to allow system operation at a suppression pool maximum temperature condition up to 121 degrees Celsius (C) /[250 degrees Fahrenheit (F)] during a beyond-design-basis event (BDBE) including the loss of onsite and offsite ac (e.g., extended station blackout (SBO)). The RCIC system is a safety system consisting of a steam turbine, pump, piping, valves, accessories, and instrumentation designed to provide sufficient reactor water inventory without ac power for at least 2 hours. Combined license (COL) applicants shall provide the analyses as part of the COL inspections, tests, analyses, and acceptance criteria (ITAAC) for the as-built facility to demonstrate that the facility has the design basis 2-hour reactor inventory capability and non-design basis 8-hour SBO capability. In addition, GEH enhanced the ACIWA subsystem design by expanding the diesel driven ACIWA pump fuel capacity and provided additional flooding protection to further ensure availability of the ACIWA subsystem under adverse conditions for an extended time up to 72 hours as described in the ABWR DCD Tier 2, Section 19.8.1.3, "Features Selected." GEH also clarified the description in the ABWR DCD Tier 2, Sections 19.8.2.3, "Selected Features" and 19.9.7, "Procedures and Training for use of AC-Independent Water Addition System," on the existing wetwell spray and spent fuel makeup capabilities that are part of the original design.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 26 of the July 20, 2012, staff letter, GEH was asked to address ABWR DCD design changes related to aspects of the NRC Fukushima Near-Term Task Force (NTTF) Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events. This recommendation was based on the NRC Commission policy at that time outlined in a staff requirements memorandum for SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A111).

Subsequently, during the pending draft mitigation of beyond-design-basis events (MBDBE) rule (10 CFR § 50.155, "Mitigation of beyond-design-basis events"), the

Commission decided not to impose mitigation strategies requirements on DCs.[8] The final rule was published in the *Federal Register* on August 9, 2019 (84 FR 39684) and became effective September 9, 2019.

Therefore, In a letter dated January 23, 2017 (ADAMS Accession No. ML17025A386), GEH submitted a revised response which removed references to the NTTF Recommendation 4.2 based on SECY-12-0025 and described the design changes in the renewal application that it had retained related to Item No. 26, as proposed design enhancements, to the ABWR certified design including the addition of an ACIWA mode to Loop B of the RHR system. As a result, future ABWR COL applicants could use these design enhancements to satisfy the MBDBE rule requirements.

These changes do not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), these design changes are "amendments," as this term is defined in Chapter 1 of this supplemental FSER, and will correspondingly be evaluated using the regulations in effect at renewal. The applicable regulatory requirements for evaluating the proposed DCD design changes to add an ACIWA subsystem to Loop B of the RHR system and related changes as discussed above are as follows:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 1, "Quality Standards and Records," as to the requirement that structures, systems, and components be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.
- GDC 34, "Residual Heat Removal," as it relates to the ABWR RHR system, which requires the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are not exceeded. Compliance with GDC 34 enhances plant safety by providing assurance that decay and RHR system functions will be accomplished and the reactor coolant system pressure boundary and fuel cladding integrity will be maintained, thereby minimizing the potential for the release of fission products to the environment.
- 10 CFR § 50.55a, "Codes and Standards," as to the establishment of minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of components of boiling and pressurized water reactor nuclear power plants by requiring conformance with appropriate editions and addenda of industry codes and standards incorporated by reference in 10 CFR§ 50.55a.
- 10 CFR § 52.47(b)(1), which requires that a DC 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 certified design has been constructed and will be operated in conformity with the certified design, the provisions of the Atomic Energy Act (AEA), and the NRC's regulations.

[8] In the MBDBE proposed rule regulatory analysis (ADAMS Accession No. ML15266A133), the Commission proposed to not make the MBDBE proposed rule applicable to existing DCs, which included the ABWR, because "[t]he issues that may be resolved in a DC and accorded issue finality may not include operational matters, such as the elements of the [MBDBE] proposed rule."

The staff used the following guidance to determine if the design of systems and components meets the regulatory requirements given above:

- Regulatory Guide (RG) 1.26, Revision 5, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," February 2017.
- RG 1.29, Revision 5, "Seismic Design Classification for Nuclear Power Plants," July 2016.

*5.4.7.1.1.10.2 Summary of Technical Information*

In its January 23, 2017, letter, GEH provided in Enclosure 1, Table 1, the enhanced design features that it had retained as part of its response to the staff Item No. 26 request regarding mitigation strategies for beyond-design-basis external events. This SER evaluates Items 1, 2, 3 and 5 of Table 1 of the January 23, 2017, letter enclosure which included the following DCD Changes:

- ACIWA subsystem enhancements (Item 1) described in DCD Tier 1, Section 2.4.1, and Figure 2.4.1.b, Tier 2, Table 1AA-2, Table 3.2-1, Table 3.9-8, Attachments 3MA.2.2 and 3MA.2.3, Sections 5.4.7.1, 5.4.7.1.1.10, 5.4.7.1.1.10.4, and Section 5.4.7.2.6, Figure 5.4.-10 SH 5 and 7;
- The diesel driven ACIWA pump fuel capacity (Item 2) described in DCD Tier 2, Section 19.8.1.3;
- The RCIC operation to 121°C/ [250°F] (Item 3) described in DCD Tier 1, Sections 2.4.4, and Table 2.4.4, DCD Tier 2, Section 5.4.6.1.1.1, and Table 5.4-2, "Design Parameters for RCIC System Components," during BDBEs; and
- The enhanced functional description for the wetwell and SFP markup capabilities described in DCD Tier 2, Section 19.8.2.3, Tables 19.8-2 and 19.8-7, and Section 19.9.

In ABWR DCD Tier 1, Section 2.4.4, the applicant revised the DCD to state that the RCIC system is capable of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool, and a suction temperature of 121°C (250°F) during BDBEs (e.g., extended SBO). To account for the higher operating temperature 121°C (250°F) during BDBEs, the applicant revised DCD Tier 2, Table 5.4-2, for the acceptable range of the RCIC pump operating water temperature to add "40°C to 121°C during BDBEs (e.g., extended SBO).

*5.4.7.1.1.10.3 Technical Evaluation*

The NRC staff reviewed ABWR DCD, Revision 7, to verify that the provisions for the ACIWA subsystem valve design, qualification (functional, environmental, and seismic), and in-service testing (IST) programs are performed in accordance with the applicable regulations, and that DCD Tier 2, Table 3.2-1, "Classification Summary," specifies the required classification for the safety-related portion of the ACIWA subsystem as Safety Class 2, Quality Group B, and seismic Category I, with 10 CFR Part 50, Appendix B, quality assurance requirements. The staff also reviewed the specific design for the additional ACIWA subsystem and its isolation valve classification for consistency with RG 1.26 and RG 1.29 and that the classification is in accordance with the requirements of 10 CFR § 50.55a.

DCD Tier 2, Table 3.9-8, "Inservice Testing Safety-Related Pump and Valves," specifies the IST provisions for valves RHR-F101B/C and RHR-F102B/C as Safety Class 2, Category B active valves, and an exercise frequency of every 3 months. The staff determined that the exercise frequency for valves RHR-F101B/C and RHR-F102B/C is consistent with the requirements in 10 CFR § 50.55a and ASME/ANSI OMa-1988 Addenda to ASME/ANSI Standard OM-1987, "Operation and Maintenance of Nuclear Power Plants."

In DCD Tier 1, Section 2.4.4, the applicant revised the DCD to state that the RCIC system is capable of injecting sufficient water to the vessel to maintain core cooling with its suction aligned to the suppression pool, and a suction temperature of 121°C [250°F] during postulated BDBEs (e.g., extended SBO). To account for the potential higher operating temperature 121°C [250°F] during BDBEs, the applicant also revised DCD Tier 2, Table 5.4-2, "Design Parameters for RCIC System Components," for the acceptable range of the RCIC pump operating water temperature to add 40°C up to a maximum wetwell temperature of 121°C [250°F] in the event of a BDBE (e.g., extended SBO).

During a postulated BDBE (e.g., extended SBO), the RCIC pump performance requirements could exceed their original safety-related design and performance specifications. Therefore, the applicant added ITAAC No. 11 in ABWR DCD Tier 1, Table 2.4.4, "Reactor Core Isolation Cooling System," with the design commitment that the RCIC system has the capability of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool, and a suction temperature of up to 121°C [250°F] during postulated BDBEs (e.g., extended SBO). ITAAC No. 11 also states that analyses will be performed of the as-built RCIC system to assess the system capability with 121°C [250°F] water at the pump suction.

An ABWR COL applicant will address operation of the RCIC system as described in ITAAC No. 11 of DCD Tier 1, Table 2.4.4 and the ACIWA subsystem for vessel injection, drywell or wetwell spray operation, and SFP makeup as described in DCD Tier 2, Section 19.8.2.3. The enhanced DCD descriptions of these modes of operation will enable an applicant to develop the necessary procedures for operation in any of these modes for preventing and mitigating severe accidents. The ACIWA subsystem valves are shown in DCD Tier 2, Figure 5.4-10. The diesel fire pump will start automatically when the ACIWA subsystem is properly aligned. If the normal firewater system water supply is unavailable, the alternate water supply can be made available by opening the manual valve between the diesel driven fire pump and the alternate water supply. This valve is shown in DCD Tier 2, Figure 9.5-4, "Fire Protection Water Supply System." If it is necessary to use a fire truck, valve F103B/C must be opened, as described in DCD Tier 2, Section 19K.11.5, "AC-Independent Water Addition (Firewater) System," in addition to operation of the valves discussed above for ACIWA subsystem operation. The valve for operation of the ACIWA subsystem using the fire truck is also shown in DCD Tier 2, Figure 5.4-10. All the valves required for ACIWA subsystem operation are manually operable so that in the event of a BDBE (e.g., extended SBO), the system can be aligned for use as necessary.

The NRC staff reviewed and verified that ABWR DCD, Revision 6, includes the following provisions for the design, qualification, and IST programs for the ACIWA subsystem valves.  DCD Tier 2, Section 3.9.3, “ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures,” specifies design provisions for Class 1, 2, and 3 valves in accordance with ASME Boiler and Pressure Vessel Code (BPV Code), Section III requirements.  DCD Tier 2, Sections 3.9.3, 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment,” and DCD Tier 2, Section 3.11, “Environmental Qualification of Safety-Related Mechanical and Electrical Equipment,” specify provisions for functional, seismic, and environmental qualification for the ACIWA subsystem valves.  DCD Tier 2, Section 3.9.6, “Testing of Pumps and Valves,” specifies IST to be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI Standard OM-1987.

The NRC staff notes that valves RHR-F790B/C, RHR-F591B/C and RHR-F592B/C (i.e., vent, drain, instrument, and test valves) are exempt from the ASME OM IST program by code due to size and function.  In addition, the NRC regulations in 10 CFR § 50.55a(f)(4) require a COL holder for an ABWR nuclear power plant to update its IST program to the latest ASME OM Code incorporated by reference in 10 CFR § 50.55a a specific time period before fuel load for the initial 120-month IST program interval.

The staff reviewed the design changes as described in the GEH January 23, 2017, letter, Enclosure 1, Table 1, Items 1, 2, 3, and 5 and determined them to be acceptable design enhancements that meet the applicable regulations for the following reasons:

- The proposed design enhancements in Item No. 1 of the GEH January 23, 2017, letter, provide an additional ACIWA subsystem to Loop B of the RHR system, and add the component designation “C” for the existing ACIWA subsystem components in Loop C of the RHR, which provides additional safe-shutdown capabilities for the ABWR and continue to meet GDC 34.
- The proposed design enhancements in Item No. 2 provide additional requirements to ensure the availability of the ACIWA subsystem under adverse conditions for up to 72 hours based on an increase of the fire diesel fuel capacity which could be used to meet requirements of the final MBDBE rule and GDC 34.
- The extension of the RCIC operating temperature (Item No. 3) for beyond-design-basis operating conditions up to a maximum of 121°C [250°F] extends the capability of the RCIC during a loss of all ac power which could be used to meet requirements of the final MBDBE rule and GDC 34.
- The proposed changes in Item No. 5 provide clarification on the use of the ACIWA for wetwell spray operation and SFP makeup capabilities which allows a potential COL applicant a means to develop the applicable procedures for operations regarding the enhanced functional description for the wetwell and SFP makeup capabilities using the ACIWA subsystem with the capabilities that had already existed and would continue to meet quality assurance requirements of GDC 1.

These proposed ABWR DC Renewal design enhancements could be used by a prospective COL applicant to meet the final MBDBE rule requirements and would continue to meet all the applicable requirements as described above.

#### *5.4.7.1.1.10.4 Conclusion*

The NRC staff reviewed the proposed GEH design enhancements that were evaluated as ABWR DCD amendments as described in the GEH letter dated January 23, 2017, Enclosure 1, Table 1, Items 1, 2, 3, and 5. The staff determined them to be acceptable design changes to the ABWR DCD because the proposed additional ACIWA subsystem to Loop B of the RHR system provides additional capabilities for plant cooldown in the event of a loss of all ac power and provides additional flooding protection and diesel fuel capacity for the non-safety fire diesel to ensure the availability of the ACIWA subsystem under adverse conditions for 72 hours. Additionally, the ABWR DCD clarifications as outlined in Item No. 5 of the January 23, 2017, applicant letter for wetwell spray operation and SFP makeup enhance a potential COL applicant's ability to develop the necessary operating procedures that could be used to meet the requirements of the final MBDBE rule. In addition, since the safety-related RHR system that interfaces with the proposed additional ACIWA subsystem will not be affected by this amendment due to the isolation valves testing, alignment, and safety design, the RHR system will function as previously designed with the additional enhancements of operation and additional flexibility such that the GDC 34 requirements are maintained and/or enhanced, and therefore these design enhancements are acceptable.

Since the safety-related portion of the ACIWA subsystem isolation valves that interface with the safety-related RHR system are classified as Safety Class 2, Quality Group B, and seismic Category I, with 10 CFR Part 50, Appendix B, quality assurance requirements the additional isolation valves added for Loop B are acceptable. These manual valves are designed to separate the safety-related portions of the RHR system from the non-safety portions of the fire protection system. Additional isolation valves for this function were added as part of the additional ACIWA subsystem added to the RHR system Loop B. These additional ACIWA subsystem isolation valves for Loop B are the same as previously used for the re-designated Loop C valves and the design and classifications are consistent with RG 1.26 and RG 1.29, and are therefore acceptable.

DCD Tier 2, Table 3.9-8, specifies the IST provisions for valves RHR-F101B/C and RHR-F102B/C as Safety Class 2, Category B active valves, and an exercise frequency of every 3 months. The exercise frequency for valves RHR-F101B/C and RHR-F102B/C is consistent with the requirements in 10 CFR § 50.55a, and ASME/ANSI OMa-1988 Addenda to ASME/ANSI Standard OM-1987. A COL applicant would use the latest version of the ASME OM Code incorporated by reference in 10 CFR § 50.55a a specific time period before fuel load for the initial 120-month IST program interval for the development of its IST program. Therefore, the ABWR DCD specified IST provisions are acceptable.

The NRC staff finds that the testing and inspection requirements in proposed ABWR DCD ITAAC No. 11 to analyze the RCIC system (including the RCIC pump) provide the necessary testing verification to ensure that the RCIC pump will operate at the pump suction water temperature up to 121°C [250°F] during BDBE conditions and meets the requirements of 10 CFR § 52.47(b)(1) to include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance of RCIC operation in a beyond design basis condition. Therefore, this proposed testing and inspection requirements are acceptable.

Based on the above, the NRC staff finds the ACIWA subsystem addition and the related design enhancements to be acceptable. The design enhancements meet the applicable regulations as stated above including the valve classification and the provisions for the design, qualification (functional, environmental, and seismic), and IST programs.

### 5.4.8 Reactor Water Cleanup System

#### *5.4.8.1 Regulatory Criteria*

The ABWR DCD, Revision 7, includes changes to address three major areas as defined in interim staff guidance (ISG)-019, DC/COL-ISG-019, "Review of Evaluation to Address Gas Accumulation issues in Safety Related Systems and Systems Important to Safety," issued September 16, 2011 (ADAMS Accession No. ML111110572): (1) identification of potential gas accumulation locations and intrusion mechanisms, (2) addition of inspection, tests, analyses, and acceptance criteria (ITAAC) to confirm identification and prevention measures, and (3) development of procedures for surveillance and venting. The changes in Chapter 5 of the DCD provide features that mitigate the possible accumulation of gases in safety-related systems and other important piping systems.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR design certification. The applicant was requested in Item No. 10 to address the three major review areas of DC/COL-ISG-019. In a follow-up letter dated July 7, 2015 (ADAMS Accession No. ML15188A255), GEH proposed a revision that included changes to important piping systems such as reactor water cleanup system (RWCS) to include a high point vent at the reactor pressure vessel (RPV) head spray line to the main steam line to avoid accumulation of hydrogen generated during normal reactor operation by radiolysis. To address further staff concerns that were discussed in a public teleconference on August 13, 2015 (ADAMS Accession No. ML15230A204), followed by a letter dated September 21, 2015 (ADAMS Accession No. ML15267A060), GEH provided supplemental information to clarify the proposed change to the RWCU system vent with respect to DC/COL-ISG-019. The new vent line does not introduce a high pressure to low pressure interface and therefore does not impact the inter-system loss-of-coolant accident information in the ABWR DCD. Furthermore, at the request of the staff during a public teleconference dated June 14, 2018, GEH proposed an additional ABWR DCD revision in a letter dated June 22, 2018 (ADAMS Accession No. ML18173A050), to add a combined license (COL) Information Item that would specifically address gas accumulation in the emergency core cooling systems (ECCS) pump suction line piping including an analysis of the ECCS suction piping to determine potential gas accumulation locations and gas intrusion mechanisms.

These changes do not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), these design changes are "amendments," as this term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations in effect at renewal. The applicable regulatory requirements for evaluating the ABWR DCD modification to address gas accumulation are as follows:

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 34, "Residual Heat Removal," as it relates to the ABWR Residual Heat Removal (RHR) system, requires the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are

not exceeded. Compliance with GDC 34 enhances plant safety by providing assurance that decay and RHR will be accomplished and the reactor coolant system (RCS) pressure boundary and fuel cladding integrity will be maintained, thereby minimizing the potential for the release of fission products to the environment.

2. GDC 35, "Emergency Core Cooling," as it relates to the ECCS system, requires the capability to provide an abundance of core cooling to transfer heat from the core at a rate such that fuel and clad damage changes in core geometry will be such that the core remains amenable to effective core cooling and clad metal-water reaction is limited to a negligible amount. Compliance with GDC 35 requires that an ECCS be provided that is capable of transferring heat from the reactor core, following a loss of reactor coolant, at a rate sufficient to ensure that the core remains in a coolable geometry, and that the calculated cladding oxidation and hydrogen generation meet the specified performance criteria.

3. TMI Action Plan- Item II.B.1 of NUREG–0737, "Clarification of TMI Action Plan Requirements," issued November 1980 (TMI Action Plan), equivalent to 10 CFR § 50.34(f)(2)(vi) for applicants subject to 10 CFR § 50.34(f), requires that reactor coolant system vessel head high-point vents be provided with remote operation from the control room with valve position indication. In addition, their operation shall not lead to an unacceptable increase in the probability of a loss-of-coolant accident or an unacceptable challenge to containment integrity.

*5.4.8.2 Summary of Technical Information*

DCD Tier 1, Section 2.6.1 includes an ITAAC requirement to inspect and confirm that the as-built RWCS high-point vent line to the RPV head spray line has the proper slope consistent with the design configurations. The applicant added the following ITAAC text to DCD Tier 1, Table 2.6.1:

| **Inspections, Tests, Analyses and Acceptance Criteria** | | |
|---|---|---|
| **Design Commitments** | **Inspections, Tests, Analyses** | **Acceptance Criteria** |
| 7. RPV Head Spray line will have a high point vent line with the proper slope to prevent buildup of Hydrogen Gas during operation. | 7. Inspections will be performed on the as built CUW piping to confirm proper elevation and slope. | 7. RPV Head Spray line will have a high point vent line with the proper slope to prevent buildup of Hydrogen Gas during operation. |

- DCD Tier 2, Section 5.4.8 includes the following text:
  A vent line down to the main steam line is provided at the high point of the RPV head spray line in order to avoid accumulation of hydrogen generated by radiolysis of reactor water during normal reactor operation.

- DCD Tier 2, Section 5.4.15 includes the following COL Information Item:
  The COL applicant shall develop periodic (monthly) surveillance procedures to ensure the Main Steam Equalizing Valve and the Main Steam Drain Valve are opened for short durations to vent any potential accumulation of hydrogen in the main steam vent and equalizing lines.

DCD Tier 2, Table 1.9-1 includes the COL Information Item above. DCD Tier 2, Figures 5.1-3 and 5.4-12, piping and instrumentation diagram (P&ID), include the new vent line modification to the Main Steam and RWCU Head Spray piping.

In the letter dated June 22, 2018, GEH proposed a COL Information Item to address gas accumulation in the ECCS pump suction line piping regarding potential gas accumulation locations and gas intrusion mechanisms.

- DCD Tier 2, Section 5.4.15 include the following COL Information Item:
  The COL applicant shall perform an analysis of the ECCS pump suction piping configuration to determine potential gas accumulation locations and gas intrusion mechanisms. In addition, the COL applicant shall address the potential for gas accumulation in ECCS on a programmatic basis that includes verification of adequate vents and other design features to prevent or mitigate gas accumulation in the pump suction line.

In the ABWR DCD Revision 7, the applicant has updated the DCD Tier 2, Table 1.9-1 with the COL Information Item discussed above.

*5.4.8.3 Technical Evaluation*

Experience from operating plants indicates that gas accumulation in ECCS and systems important to safety may render the system inoperable during a transient event. Prior to 2005, there have been at least five gas accumulation events of GE designed reactor plants that resulted in an ECCS or system important to safety being declared inoperable. Gas accumulation is known to cause water hammer, gas binding in pumps, and inadvertent relief valve actuation that may damage pumps, valves, piping, and supports. The NRC issued DC/COL-ISG-019 to provide guidance regarding safety-related systems to supplement NUREG–00800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” and other standard review plans for systems important to safety because they did not include specific concerns and guidance to the extent covered in Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems,” dated January 11, 2008, (ADAMS Accession No. ML072910759).

In support of the piping vent line configuration changes, the applicant included an ITAAC in DCD Tier 1, Table 2.6.1, to inspect and confirm that the as-built RWCU System vent piping slopes is consistent with detailed design drawings. In addition, the staff noted that the changes to the vent line include one isolation valve that connects the head spray line to the main steam line from where the accumulated gas is vented through two valves in series. The three valves are controlled from the control room, and valve position is indicated. The staff also noted that having these valves in series satisfies the TMI action item requirement that at least two vent valves must be in series to minimize the challenges to the ECCS from an inadvertent opening of a new or existing vent line.

In addition, GEH added a COL Information Item to develop periodic surveillance procedures to ensure the main steam equalizing valve and the main steam drain valve are opened for short durations to vent any potential accumulation of hydrogen in the main steam vent and equalizing lines.

The staff evaluated the configuration changes described above and finds that the changes comply with TMI Action Plan Item II.B.1, as required by 10 CFR § 50.34(f)(2)(vi), in removing potential hydrogen that may adversely affect core cooling. The staff confirmed the changes are reflected in the ABWR DCD, Revision 7.

While GEH proposed the addition of a high point vent and the main steam vent changes that satisfy the TMI action item as described in DCD Revision 6, they did not fully address DC/COL-ISG-019 guidance and, therefore, the requirements of GDC 34 and 35 with respect to gas accumulation in safety-related systems and systems important to safety. Therefore, in its letter dated June 22, 2018, GEH proposed a COL Information Item to address gas accumulation in the ECCS pump suction line piping by means of an analysis to identify the potential gas accumulation locations and gas intrusion mechanisms. In addition, a COL applicant referencing this design would need to address the potential for gas accumulation in ECCS on a programmatic basis that includes verification of adequate vents and other design features to prevent or mitigate gas accumulation in the pump suction line. The staff evaluated the ECCS subsystems piping configuration to determine that the COL Information Item sufficiently addresses the DC/COL-ISG-019 guidance as summarized below.

The ABWR ECCS consists of the following subsystems: (1) high pressure core flooder (HPCF), (2) low pressure flooder (LPFL) Mode of the RHR System, (3) reactor core isolation cooling (RCIC) system, and (4) automatic depressurization system (ADS). ADS is not considered in this evaluation because the gas accumulation is not a factor for a system composed of reactor safety relief valves (SRVs). The remaining ECCS subsystems are designed to maintain the suction piping line water filled during normal operations.

The HPCF subsystem is designed with two independent loops that take their primary suction from the condensate storage tank (CST) and secondary suction from the containment suppression pool. The HPCF pumps are located at an elevation which is below the water level of the suppression pool. This assures the pump suction line remains full. Also, for each loop, a full flow line is provided with discharge to the suppression pool to allow for a full flow test of the system during normal operation. The ABWR technical specifications specify a periodic full flow system functional test on a 92-day basis. The COL Information Item added by the applicant directs a COL applicant to perform an analysis to determine potential gas accumulation locations and gas intrusion mechanisms.

The staff finds that the HPCF design in the ABWR DCD is acceptable because the suction piping line is configured below the elevation level of the makeup sources, the suction piping line is periodically purged during the functional test, and the COL Information Item directs a COL applicant to address the suction piping to ensure consistency with guidance in DC/COL-ISG-019.

The RHR system has a LPFL subsystem mode that pumps water from the suppression pool into the reactor vessel at low reactor pressure. During normal plant operation, the RHR loops are in a standby condition with the RHR pumps not running. The RHR system is designed to have the pumps start and deliver water into the reactor vessel within 36 seconds after receipt of the low-pressure permissive signal following system initiation. Any gas accumulation in the suction line may delay the injection beyond 36 seconds, which may impact and invalidate the transient analysis. Therefore, the suction line of the RHR design includes water leg pumps (line fill pumps) which are normally running to keep the pump discharge lines filled while the RHR system is in standby mode.

However, operating plant experience has shown that the water leg pumps may become air bound unable to perform their intended function; thus, gas accumulation may occur during normal power operation. The proposed GEH COL Information Item from the letter dated June 22, 2018, directs a COL applicant to address the potential for gas accumulation in the ECCS on a programmatic basis that includes other design features to prevent or mitigate gas accumulation in the pump suction line. The staff evaluated the applicant's changes and finds that the design of the RHR system in the ABWR DCD, Revision 7, is acceptable because the water leg pumps are designed to prevent gas accumulation in the discharge line piping, the periodic functional test provides purging of the suction line, and the COL Information Item directs a COL applicant to evaluate the suction piping to ensure the design satisfies the guidance in DC/COL-ISG-019.

The RCIC system is designed to provide makeup water from the CST or the suppression pool to the reactor vessel during a reactor shutdown in which feedwater flow is not available. The system is started automatically on a low reactor water level signal or manually by the operator. Also, a design flow functional test of the RCIC system is performed periodically during normal plant operation by drawing suction from the suppression pool and discharging through a full flow test return line to the suppression pool. This test is performed to assure the system design flow and head requirements are attained within 30 seconds to support the transient analysis. During normal plant operation, the RCIC is in standby mode with the pump suction line kept filled. The flow test has the capability of removing any potential gas that may have accumulated during the 92-day testing interval as specified in the technical specifications. The COL Information Item directs a COL applicant to perform an analysis to determine potential gas accumulation locations and gas intrusion mechanisms and the necessity for additional venting and filling.

In regard to gas accumulation, the staff evaluated the RCIC design and finds that it is acceptable because the measures undertaken in the design prevent potential gas accumulation including the COL Information Item that directs a COL applicant to address the guidance in DC/COL-ISG-019 and therefore, the requirements of GDC 34 and 35 are satisfied for safety related and important to safety systems.

In summary, the staff determined that the ECCS system conforms with the guidance in DC/COL-ISG-019 because: (1) the HPCF, RCIC and RHR subsystem suction piping is below the elevation of the makeup sources, (2) the RHR LPFL subsystem suction piping has water leg pumps that maintain the discharge piping water filled, (3) ECCS subsystems are functionally tested, which also allows the purging of the suction piping, (4) the discharge piping is periodically vented and filled as specified in the technical specifications on a 92-day interval, and (5) the COL Information Item directs a COL applicant to address the analysis to be conducted to determine the necessity for additional venting and filling. Therefore, the staff finds the applicant's ABWR DCD, Revision 7, changes acceptable.

The staff confirmed that the applicant provided the requested COL information regarding ECCS gas accumulation in the ABWR DCD, Revision 7, which incorporates the changes described in the applicant’s letter dated June 22, 2018. Therefore, Confirmatory Item 5.4.8-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *5.4.8.4 Conclusion*

The staff reviewed the changes for renewal of the ABWR DC as described in the DCD Tier 1 and Tier 2 sections of ABWR DCD, Revision 7, that address conformance with DC/COL-ISG-019 and TMI Action Plan Item II.B.1. Based on the staff’s technical evaluation described in this FSER supplement, the staff found that the changes meet the guidance specified in DC/COL-ISG-019 to reduce gas accumulation in safety-related systems and systems important to safety. Regarding the TMI Action Plan, the changes add the capability of removing hydrogen from the reactor vessel head with high-point vents remotely operated from the control room. The staff finds that the changes comply with 10 CFR § 50.34(f)(2)(vi), meet the guidance specified in DC/COL-ISG-019 and TMI Action Plan Item II.B.1, and do not alter the safety findings made in NUREG–1503 and its Supplement 1, the staff FSER for the original ABWR DC. Therefore, the staff concludes that the amendments to the ABWR DCD associated with the design changes outlined above meet the requirements of GDC 34 and GDC 35 and are acceptable.

# 6 ENGINEERED SAFETY FEATURES

### *6.2.1.3 Short-Term Pressure Response*

#### *6.2.1.3.1 Regulatory Criteria*

The applicant for the ABWR DC renewal, completed design changes to the certified ABWR DCD in Revision 7, after identifying an error in the containment peak pressure analysis as discussed in a letter from GEH dated June 8, 2009 (ADAMS Accession No. ML100640164). In Enclosure 1 of the letter dated December 7, 2010, transmitting its application to renew the ABWR DC (ADAMS Accession No. ML110040176), the applicant stated, in part, the following:

> the containment peak pressure re-analysis complies with NRC regulations that were in place at the time of certification, as required by 10 CFR 52.59(a), the amendment also complies with current applicable NRC regulations. GEH expects that the applicable regulations will remain the same during the NRC review of the application. However, if the NRC amends those regulations during the time period of its review, GEH will review such amendments to determine if any further changes are necessary.

The staff assessed the design changes associated with the containment peak pressure reanalysis and determined that some of the changes would meet the criteria for modifications while others would be identified as amendments, as these terms are defined in Chapter 1 of this FSER supplement. However, due to the interrelationship of the design changes, the staff decided to treat all the changes as “amendments” to the certified design and will correspondingly evaluate the changes using the regulations applicable and in effect at renewal. GEH’s statement above regarding compliance with current regulations supports this decision. In addition, the staff determined that the pertinent requirements in current regulations and associated staff guidance for the review of the changes are not substantially different than the regulations and associated guidance in effect at the time of the original ABWR DC. Therefore, by conducting the review against current regulations, the staff’s evaluation also supports a finding of compliance with the applicable regulations in effect at initial certification.

The NRC staff’s requirements for its review are specified in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” (GDC) 16, “Containment Design,” and GDC 50, “Containment design basis,” as they relate to the containment and its associated systems being able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. In NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments," Revision 7, issued March 2007, provides guidance on acceptable analytical models for calculating the containment peak pressure and temperature.

#### *6.2.1.3.2 Summary of Technical Information*

In the ABWR DCD, Revision 6 submittal, GEH included the following changes from the original ABWR certification, incorporating changes contained in GEH’s response to

RAI 06.02.01.01.C-1, Revision 1 dated August 11, 2015 (ADAMS Accession No. ML14239A137):

- a change in decay heat curves assumed for the long-term containment analysis from nominal values in the 1979 version of the American National Standards Institute (ANSI)/American Nuclear Society (ANS)–5.1, "Decay Heat Power in Light Water Reactors" to the 1994 version with a two standard deviation uncertainty on decay heat
- containment vent system modeling changes to include the drywell connecting vent (DCV) loss coefficients to correct the modeling of horizontal vents
- the feedwater line break (FWLB) flow changes to remove the initial 3.75-second inventory depletion period in the original DCD Tier 2, Figure 6.2-3
- a change in the suppression pool water level assumed for the long-term containment response analysis from 7 meters (equivalent to a volume of 3,580 cubic meters) to 6.9 meters (equivalent to a volume of 3,455 cubic meters)
- a change in the residual heat removal system (RHR) heat exchanger overall heat transfer coefficient assumed for the long-term containment response from $3.7x10^5$ watts per degree Celsius (W/°C) to 4.27x105 W/°C (an approximately 15 percent (%) increase)
- Wetwell design temperature change from 104°C to 124°C
- negative pressure design evaluation changes including (1) eliminating analyses for events with inadvertent initiation of containment (drywell/wetwell) spray during normal operation, (2) taking credit for heating of emergency core cooling system (ECCS) flow in the reactor pressure vessel before being discharged into the drywell, and (3) using the GEH SUPERHEX computer code instead of the previous analyses, which "used a series of end-point calculations to generate a set of conditions that produces a bounding prediction of the peak negative [wetwell to reactor building] differential pressure"

In Supplements 1 and 2 of the applicant's response to RAI 06.02.01.01.C-1, Revision 1 dated May 6, 2016 (ADAMS Accession No. ML16127A032) and Revision 2 dated June 22, 2016 (ADAMS Accession No. ML16174A179), respectively, the applicant made changes to DCD Tier 2, including the following as ABWR DCD, Revision 6, markups:

(1) adding text in DCD Tier 2, Section 5.4.7.3.2, "Worst Case Transient," to state that "[t]he normal shutdown condition is used to establish the limiting heat exchanger capacity and is evaluated in Appendix 5B.3"

(2) replacing text in DCD Tier 2, Section 5.4.7.3.2, where rather than stating that RHR heat exchanger size was established to limit the suppression pool peak temperature to 97°C, the text will instead state that the heat exchanger size will also support the safety function of limiting suppression pool peak temperature to 97°C

(3) In DCD Tier 2, Table 6.2-2, "Containment Design Parameters": the vent loss coefficient (VLC) is changed from 2.5 – 5.0 to 4.2 – 6.7 and a footnote is added to the table to state that the overall vent system loss coefficient includes a contribution from flow loss coefficient of 1.7 for DCV

### *6.2.1.3.3 Technical Evaluation*

The staff reviewed the final ABWR DCD, Revision 7, changes in DCD Tier 2, Sections 6.2.1 and 6.2.2 to determine compliance with GDC 16 and 50, using the guidance in SRP Section 6.2.1.1.C, Revision 7, issued March 2007. The staff determined that additional information was needed to complete its review and issued RAI 06.02.01.01.C-1, dated April 24, 2014 (ADAMS Accession No. ML14114A566). GEH responded in a letter dated August 27, 2014 (ADAMS Accession No. ML14239A137), which it revised and replaced by the letter dated August 11, 2015. GEH supplemented its response further in the letter dated June 22, 2016.

Enclosure 1 of the DC renewal application dated December 7, 2010, the applicant made DCD changes to correct the containment peak pressure analysis to reflect a more limiting line break that GEH identified and discussed in the letter dated June 8, 2009. The limiting line breaks for the short-term accident response did not change from the certified design to the revised design. However, for the long-term accident response, revisions to FWLB analysis resulted in a change to the drywell peak pressure and revisions to the main steamline break (MSLB) analysis resulted in changes to the drywell peak temperature. The June 8, 2009, letter, refers to NEDO-33372, “Advanced Boiling Water Reactor (ABWR) Containment Analysis," which was later withdrawn from NRC topical report review by letter dated March 30, 2010 (ADAMS Accession No. ML100890313). As such, the staff was not clear about the documentation supporting the ABWR design certification renewal application changes to DCD Tier 2, Sections 6.2.1 and 6.2.2. Therefore, in Part (1) of RAI 06.02.01.01.C-1, the staff requested GEH to provide documentation supporting the containment reanalysis changes of the ABWR DCD.

In its response dated August 11, 2015, GEH stated the following:

> There are no new documents that have been issued or new references cited that were required to support the changes for the DCD revision. Although NEDO-33372 is no longer directly applicable to the ABWR for the reasons discussed above, there is certain information that remains applicable to the ABWR renewal application. Therefore, rather than revise NEDO-33372, the information is proposed to be included in the ABWR DCD.

GEH’s response identified two major and four minor changes associated with containment analysis. Major changes were associated with the decay heat used for the long-term containment analyses and modeling of the containment vent system. Minor changes were associated with FWLB flow, suppression pool volume margin, the overall heat transfer coefficient for the RHR heat exchanger, and wetwell design temperature.

The original certified ABWR DCD long-term containment analysis was based on nominal ANSI/ANS-5.1 (1979) decay heat curves. GEH determined that additional actinides and activation products not accounted for in the ANSI/ANS-5.1 (1979) standard, affect the decay heat curves. Therefore, in the revised DCD Tier 2, Section 6.2, “Containment Systems,” for long-term containment analysis, GEH used ANSI/ANS-5.1 (1994), which includes contributions from additional actinides and activation products. In addition, GEH conservatively used a two standard deviation uncertainty on decay heat when performing the revised long-term containment analysis.

The staff finds that using the ANSI/ANS-5.1 (1994) decay heat model with a two standard deviation uncertainty for the long-term containment analysis is acceptable since the addition of decay heat from actinide decay and activation products is conservative for containment pressure and temperature analysis.

In the applicant's RAI response dated August 11, 2015, GEH stated the following about the changes to the containment vents model:

> In the containment analysis for the certified ABWR DCD, the main vent system model did not capture some of the key features that impact the short-term containment response and thus the pool swell loads. The model for DCD Revision 4 did not properly simulate the horizontal vent portion of the vent system and consequently incorrectly modeled the vent clearing time. These deficiencies are the major contributor to the difference between the previous certified ABWR and the ABWR revised containment analysis results.
>
> The revised ABWR containment analysis correctly models the horizontal vents and was performed with DCV loss coefficients included. The total DCV loss coefficient is based on a summation of losses. The entrance loss coefficient accounts for the presence of the biological shield wall that is next to the upper drywell entrance to the DCV. The flow loss coefficient accounts for trash racks at the entrance to the vents to block insulation from entering the vents and flowing into the suppression pool. The friction loss through the DCV is the maximum pressure loss coefficient due to piping, cabling and supports routed in the DCV. The exit loss coefficient can be neglected since each DCV is directly above a Drywell-Wetwell (DW-WW) vertical vent. These flow losses are then summed and included in the containment analysis model for the DCV.
>
> The dimensions of the horizontal vents were included in the revised analysis and confirmation of the vent clearing was performed to ensure the revised model was correct. These modifications were the major contributors to the revised analysis results for the wetwell pressure and drywell-to-wetwell differential pressures.

GEH needed to change the containment vents model to correct self-identified errors in the containment analysis. The staff finds that the above features, which were missing in the containment analysis for the certified ABWR DCD by error, were needed to correctly model the GEH ABWR design, and therefore, determines that these modeling changes are acceptable.

DCD Tier 2, Table 6.2-2 lists the VLC range between 2.5 to 3.5. The applicant changed the VLC range to between 2.5 to 5.0. GEH cited NEDO-33372 and indicated that the applicable information was extracted from this document and put into the DCD. The staff noted that NEDO-33372 lists the VLC as between 2.5 to 3.5, which is different from the range of values provided in the ABWR DCD, Revision 5, specifically the upper limit. Therefore, in a public teleconference on April 6, 2016, the NRC staff requested GEH to clarify this difference.

In the applicant's response to RAI 06.02.01.01.C-1, Revision 1, Supplement 2, dated June 22, 2016, GEH reiterated that it does not intend for NEDO-33372 to be part of the licensing basis for ABWR DC renewal and the ABWR DCD will contain all pertinent content and that the range of VLC values shown in the markup for DCD Tier 2,

Table 6.2-2 that was included in NEDO-33372 does show a VLC range of 2.5 – 5.0. The original upper end value of 3.5 is shown crossed out in the markup.

The original range of 2.5 - 3.5 was first developed for use with the GEH Mark III Containment Pressure and Temperature (M3CPT) code for analyses of the Mark III short-term containment response. It was then applied in the ABWR M3CPT analyses due to the similarity in the Mark III and ABWR horizontal vent system geometry. A subsequent evaluation updated the range of VLCs for Mark III M3CPT analysis to 2.5 - 5.0. The revised values were then also applied to the ABWR M3CPT containment analysis. The values shown in DCD Tier 2, Table 6.2.2 (2.5 - 5.0) only included the losses associated with the ABWR vent system. It did not include or identify a 1.7 loss coefficient adder to the values shown in DCD Tier 2, Table 6.2-2 that was applied to account for flow losses associated with the DCV that connects the upper drywell to the vent system. The applicant provided a markup for the ABWR DCD, Revision 6, identifying the range of overall VLCs used for the analyses for the ABWR DCD that includes the 1.7 loss coefficient adder (4.2 - 6.7).

The applicant provided the ABWR DCD, Revision 7, value for VLC as (4.2 - 6.7), that the staff found conservative and therefore acceptable, as incorporated from the applicant's response to RAI 06.02.01.01.C-1. Therefore, Confirmatory Item 6.2.3.1-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

The FWLB flow change was to increase the 116 percent nuclear boiler rated (NBR) flow from the balance of plant side during the initial 3.75-second feedwater inventory depletion period to 164 percent NBR flow, as assumed for the inventory depletion period after the 3.75-second period and shown in the certified DCD Tier 2, Figure 6.2-3. The specific enthalpy of feed water flow as shown in DCD Tier 2, Figure 6.2-4 was unchanged. This increase in mass flow is conservative because it produces a higher energy flow into the containment than that used in the certified ABWR design during a FWLB, resulting in higher short-term containment peak pressures. Therefore, the staff finds that the FWLB flow change is acceptable.

In the applicant's RAI response dated August 11, 2015, GEH stated the following about the change in suppression pool volume margin:

> The water volume in the suppression pool including the vents is required to be equal to or greater than 3,580 cubic meters, as stated in the Tier 1 Section 2.14.1. The ABWR revised [long-term] containment analyses of scenarios with low initial suppression pool water level were performed with a smaller water volume (3,455 cubic meters) to ensure analysis/operational margin. This smaller volume is based on a suppression pool water level of 6.9 meters. The volume of 3,580 cubic meters is equivalent to a 7-meter water level. The technical specification for suppression pool water level (LCO 3.6.2.2) is greater than or equal to 7 meters and less than or equal to 7.1 meters. This is a very tight band to control the suppression pool water level; so additional margin (0.1 meters) has been built-in to the safety analysis. It is conservative to base the safety analysis for scenarios with a lower initial suppression pool water level based on a smaller water volume as this results in higher pool temperatures.

The staff determined that the change in the suppression pool volume margin in the safety analysis is conservative for as described by the applicant, and therefore, is acceptable.

As part of the applicant's response to RAI 06.02.01.01.C-1 dated August 11, 2015, and supplemented with the response dated May 6, 2016, GEH increased the RHR heat exchanger's heat transfer coefficient. However, cooldown rates for the ABWR are administratively controlled and are governed by technical specifications; therefore, the staff concluded that the increase in the heat transfer coefficient RHR does not affect the safety of the reactor or the containment analysis.

The staff reviewed the ABWR DCD, Revision 6, markups and confirmed that the applicant has made the appropriate changes in the ABWR DCD, Revision 7, from the response to RAI 06.02.01.01.C-1 in the May 6, 2016 supplemental letter. Therefore, Confirmatory Item 6.2.3.1-1 from the staff advanced safety evaluation report with no open items for the ABWR DC renewal is resolved and closed.

On the wetwell design temperature change, GEH's response dated August 11, 2015, stated the following:

> The certified ABWR wetwell gas space design temperature was 104°C. The containment structural analysis design value is 124°C; therefore the Tier 2 DCD wetwell chamber design temperature was revised to 124°C.

The staff finds this change acceptable because it is more protective from a safety standpoint and makes the containment structural analysis and wetwell chamber design temperatures consistent.

As described above, the staff finds that GEH's response to Part (1) of RAI 06.02.01.01.C-1 acceptable.

In DCD Tier 2, Chapter 6, Change List Item 18 (which is related to DCD Tier 2, Section 6.2.1.1.3.3.1.2) it is stated that lower drywell flooding is not modeled. The staff was not clear why lower drywell flooding was not modeled. Therefore, in Part (2) of RAI 06.02.01.01.C-1, the staff requested GEH to justify not modeling lower drywell flooding.

In the applicant's response letter dated August 11, 2015, GEH described two mechanisms causing lower drywell flooding. The first was spilling of break flow water from the upper drywell to the lower drywell through the DCV connection. GEH assumed that water, that can spill, into the lower drywell would flow into the suppression pool instead. This assumption is conservative because water that flows out from the break during suppression pool drawdown during ECCS injection will be hotter than the water in the suppression pool and adding it back to the suppression pool would heat the suppression pool water.

A second mechanism is the potential for reverse vent flow from the suppression pool to the lower drywell through the lower drywell overflow orifice connection to the vertical vent. GEH showed that extended periods of large negative DW-WW pressure gradients would not exist because of opening of wetwell-to-drywell vacuum breakers. Further DCD Tier 2, Section 6.2.1.1.10.3 states the following:

> The interconnection between the lower drywell and the wetwell is at elevation -4.55 m, [which is] 8.6 m above the floor of the suppression pool. Thus, approximately 7.2E5 kg of water must be added from outside the containment for the suppression pool to overflow into the lower drywell.

As such, reverse vent flow from the suppression pool to the lower drywell would be unlikely to occur. Therefore, the staff finds that GEH's justification for not modeling the lower drywell flooding as provided in the response to Part (2) of RAI 06.02.01.01.C-1 is acceptable.

In the ABWR DC renewal application, ABWR DCD, Revision 5, Tier 2, Change List Item 19 (which is related to DCD Tier 2, Section 6.2.1.1.3.3.1.2, "Assumptions for Long-Term Cooling Analysis"), the applicant deleted previous assumption No. 7 from the certified design and inserted an assumption stating that the structural heat sinks are credited. The previous assumption, which was deleted, stated that at 70 seconds, the feedwater specific enthalpy becomes 418.7 Joules per gram (J/g) (i.e., saturation fluid enthalpy at 100 degrees Celsius (°C). The staff finds that removing previous assumption No. 7 is acceptable because the applicant used DCD Tier 2, Figure 6.2-22, from the ABWR DCD, Revision 4, certified ABWR DCD instead, which provides a more limiting value for feedwater specific enthalpy. DCD Tier 2, Figure 6.2-22 shows that the feedwater specific enthalpy drops below 418.7 J/g only after 86 seconds.

However, the application did not provide the details for its modeling of the heat sinks. Therefore, in Part (3) of RAI 06.02.01.01.C-1, the staff asked GEH to provide this information. In the applicant's response dated August 11, 2015, GEH provided details for its modeling of the structural heat sinks in the drywell airspace, wetwell airspace, and suppression pool. The applicant has modeled heat transfer in the drywell and wetwell air space by natural convection and condensation. The applicant modeled heat transfer from the suppression pool water to the suppression pool heat sinks. The applicant's response included tables of heat sink parameters for the modeled heat sinks in the drywell airspace, wetwell airspace, and suppression pool. The applicant stated that the crediting of the heat sinks remains valid for as-built plants unless there is a change in plant dimensions. However, a COL applicant will include inputs for heat sinks in the standard form that the applicant uses to confirm inputs to the containment analysis and confirm the validity of the ABWR DCD analysis for the as-built plant. Design Commitment 4 in DCD Tier 1, and DCD Tier 1, Table 2.14.1 states that "[t]he maximum calculated pressures and temperatures for the design basis accident are less than design conditions." The discussion of inspections, tests, and analyses for this commitment states that "[a]nalyses of the design basis accident will be performed using as-built [primary containment system] data." The applicant provided tables with properties of heat sinks in response to Part (3) of RAI 06.02.01.01.C-1. The staff reviewed these properties to confirm that the applicant used correct thermal properties and correctly calculated the mass and internal thermal resistance for the heat sinks. Based on its review, the staff finds that the applicant's response to Part (3) of RAI 06.02.01.01.C-1 is acceptable.

DCD Tier 2, Chapter 6, Revision 5, Change List Item 23 (which is related to the main steamline break discussion in DCD Section 6.2.1.1.3.3.2) changed assumption (5). Assumption (5) in the certified ABWR DCD, Revision 4, stated that "MSIVs are completely closed at a conservative closing time of 5.5 seconds (0.5 seconds greater than the maximum closing time plus instrument delay), in order to maximize the break flow." ABWR DCD, Revision 5, changed the closing time to 5 seconds and eliminated the reference to 0.5 seconds delay. The staff was not clear whether the 0.5-seconds delay was included in the MSIV closing time. Therefore, in Part (4) to RAI 06.02.01.01.C-1, the staff requested GEH to clarify the MSIV closing time.

In the applicant's response dated August 11, 2015, GEH stated that the instrument delay of 0.5 seconds to begin closing the MSIVs is included in the total 5.0 second duration for MSIV closure from the start of the event. This clarifies how the instrument delay of 0.5 seconds is accounted for. The staff finds that closing the MSIVs sooner (i.e., in 5 seconds versus 5.5 seconds used in the certified ABWR DCD) is conservative because it reduces radioactive releases through MSIVs during a design basis accident. Based on its review, the staff finds that GEH's response to Part (4) of RAI 06.02.01.01.C-1 acceptable.

DCD Tier 2, Revision 5, Chapter 6, Change List Item 24 relates to changing assumptions used in short-term containment analysis in DCD Tier 2, Section 6.2.1.1.3.3.2.1. GEH deleted the following assumptions:

- Assumption 1. The vessel depressurization flow rates are calculated using the Moody's [homogeneous equilibrium model (HEM)] for the critical break flow.
- Assumption 2. The turbine stop valve closes at 0.2 second. This determines how much steam flows out of the RPV, but does not affect the inventory depletion time on the piping side.
- Assumption 4. The feedwater mass flow rate for a [main steam line] break was assumed to be 130 percent of NBR for 120 seconds. This is a standard [MSLB] containment analysis assumption based on a conservative estimate of the total available feedwater inventory and the maximum flow available from the feedwater pumps with discharge pressure equal to the [reactor pressure vessel] pressure. The feedwater enthalpy was calculated as described for the [FWLB] (Subsection 6.2.1.1.3.3.1.1) for 130 percent of NBR flow, and is shown in Figure 6.2-11.

The reason for these deletions was not clear to the staff. Therefore, in Part (5) of RAI 06.02.01.01.C-1, the staff asked GEH to explain these changes. In the applicant's response dated August 11, 2015, GEH noted that Assumption 1 was listed as an exception to the assumptions identified for the FWLB analysis in DCD Tier 2, Section 6.2.1.1.3.3.1.1. GEH deleted this assumption in DCD Tier 2, Section 6.2.1.1.3.3.2.1 because the same assumption is listed under DCD Tier 2, Section 6.2.1.1.3.3.1.1. The staff finds that deletion of Assumption 1 acceptable because the deletion was to remove a repetitive assumption in the ABWR DCD.

Concerning the deletion of Assumption 2, GEH noted that the use of the turbine stop valve closure time is not applied for the revised MSLB analysis to establish the vessel isolation time, and Assumption 5 in Section 6.2.1.1.3.3.2 states that "MSIVs are completely closed at a conservative closing time of 5 seconds in order to maximize the break flow." The staff finds GEH's deletion of Assumption 2 acceptable because it is not used for the revised analysis.

Concerning the deletion of Assumption 4, GEH stated that Assumption 4 describes feedwater injection to the vessel for the MSLB, which is not modeled in the current short-term MSLB analysis. Injecting relatively colder feedwater into the reactor pressure vessel will tend to reduce the short-term vessel pressure due to reduced steaming that in turn reduces the break flow into the containment, thereby lowering the predicted short-term MSLB containment pressure and temperature. Therefore, to produce a more conservative short-term MSLB pressure and temperature response, the applicant has not included feedwater injection in the MSLB short-term analysis. The staff finds the applicant's deletion of Assumption 4 conservative and acceptable.

As described above, the staff finds GEH's response to Part (5) of RAI 06.02.01.01.C-1 acceptable.

DCD Tier 2, Revision 5, Chapter 6, Change List Item 26 (which is related to the discussion of short-term accident response in DCD Tier 2, Section 6.2.1.1.3.3.2.3) indicates that the short-term MSLB has a more severe drywell temperature response than before as it increased from 169.7 °C in ABWR DCD, Revision 4, to 177.2 °C in ABWR DCD, Revision 5. The reason for this change was not clear to the staff. Therefore, in Part (6) to RAI 06.02.01.01.C-1, the staff requested GEH to explain.

In the applicant's response dated August 11, 2015, GEH stated the following:

> The revised analysis included corrections to the vent system modeling that had a significant impact on both the peak drywell pressure and peak drywell temperature. The length of the horizontal vent was not correctly accounted for in the original calculation. In addition, the overall flow loss coefficient for the ABWR vent system did not account for the flow losses associated with the drywell connecting vents (DCV). The corrections that were implemented in the revised calculations produced a delay in clearing of the horizontal vents and an increase in the vent flow resistance after vent clearing. These changes produced the higher values for predicted peak MSLB drywell pressure and temperature.
>
> The peak calculated MSLB drywell temperature of 177.2 °C is higher than the design limit of 171.1 °C. However, this value represents the peak predicted MSLB drywell atmosphere temperature. A review of the analysis shows that predicted drywell atmosphere temperatures are above 171.1 °C for approximately only 1 second during the early, steam break flow only phase of the MSLB. The MSLB analysis assumes level swell of the vessel liquid due to voiding, which produces a two-phase break flow mixture after two seconds into the event. Thereafter, drywell temperatures fall rapidly (see DCD Tier 2, Figure 6.2-13). The very short predicted duration of atmosphere temperature above 171.1 °C will not result in drywell structural temperatures that are above the drywell structure design limit.

The applicant corrected a self-identified error in modeling the overall flow loss coefficient for the ABWR vent system. The staff reviewed these modeling changes under Part (1) to RAI 06.02.01.01.C-1 and found them acceptable. The peak calculated MSLB drywell atmosphere temperature of 177.2 °C exceeds the drywell design limit of 171.1 °C for a 1 second duration. However, due to thermal inertia, components in the drywell structures (in particular, the upper head seals) will not have sufficient time to reach the design limit temperature during this 1 second period. Therefore, the staff finds that containment atmosphere temperature exceeding the structural design temperature in this case is acceptable. Based on its review, the staff finds GEH's response to Part (6) of RAI 06.02.01.01.C-1 is acceptable.

DCD Tier 2, Chapter 6, Revision 5, Change List Items 30 and 31 (which are related to DCD Tier 2, Section 6.2.1.1.4.1 and Section 6.2.1.1.4.2 on the negative pressure design evaluation) states that the applicant replaced each section except the first two paragraphs. The applicant did not state the reasons for the changes. Therefore, in Part (7) to RAI 06.02.01.01.C-1, the staff requested the applicant provide details justifying the changes. In the applicant's response dated August 11, 2015, GEH stated that it performed the revised calculations to provide a more accurate and realistic simulation of negative pressurization events consistent with the ABWR plant system design, plant system operation and plant operating conditions. The main changes made in the revised analysis are as follows:

- The applicant eliminated analyses for events with inadvertent initiation of containment (drywell/wetwell) spray during normal operation. As described in DCD Tier 2, Section 6.2.1.1.4, the ABWR design has features that prevent the initiation of the RHR mode of the drywell spray(s) during normal plant operation.
- The revised analyses start at time zero of the postulated loss-of-coolant accident event with normal operating conditions as the initial conditions. The analysis itself is used to predict the initial conditions prior to ECCS reflood or drywell (DW) spray initiation as opposed to using user-defined conditions at the time of ECCS reflood or spray.
- Drywell break flow rate and break flow enthalpy during periods of ECCS injection are mechanistically calculated considering the effects of ECCS injection rates, ECCS source temperature, and heatup in the vessel before discharge to the drywell.
- The revised analyses include modeling of DW spray with suction from the suppression pool. The DW spray temperature is established by the calculated exit temperature of the modeled RHR heat exchanger and accounts for the heat exchanger heat removal characteristics (heat exchanger coefficient), calculated suppression pool temperature, RHR service water temperature and containment spray flow rate.
- The new analyses include a small steamline break with DW spray operation to provide the containment negative pressure response due to operation of drywell spray in a superheated steam drywell environment, which would occur during a small steam break, and which is potentially limiting for containment negative pressure.

The following provides the staff's evaluation of the above changes:

- As stated in DCD Tier 2, Section 6.2.1.1.4, of the ABWR DCD, Revision 6, an interlock on the drywell spray injection valves that requires high drywell pressure to be present before the valves are allowed to be opened and a time delay in the logic will allow initiation of drywell spray 60 seconds after the drywell high pressure signal (2 psig) is received. In addition, the RHR system can only be manually initiated in the drywell spray mode from the main control room by two methods, both requiring two independent actions. Therefore, the staff finds that a likelihood of a spurious initiation of drywell spray during normal plant operation to be remote and the elimination of such activation from analysis to be acceptable.

- The applicant used the analysis itself, rather than user defined conditions, to establish initiation of the negative design pressure evaluation. The staff finds that this approach is less subjective, and therefore, acceptable.
- The applicant stated in its RAI response dated August 11, 2015, that "[i]n the original DCD analysis it was assumed that 100 percent of ECCS flow (including [high pressure core flooder, low pressure core flooder and reactor core isolation cooling]) is taken from the [condensate storage tank] (at 60°F) and discharged directly into the drywell without heating of the ECCS injection fluid in the vessel." Using mechanistically calculated drywell break flow rate and break-flow enthalpy during periods of ECCS injection produces a less conservative result than that provided in the certified ABWR DCD. However, the staff finds the applicant's mechanistic calculation consistent with SRP Section 6.2.1.1.C, and therefore, acceptable.
- The analysis presented in the certified ABWR DCD did not assume the operation of drywell sprays. The staff finds that the operation of drywell sprays would lower the drywell temperature and pressure by condensing steam in the drywell, which conservatively increases the DW-WW negative pressure, and therefore, is acceptable.
- As stated under Item 4 above, operation of drywell sprays in a steam environment would lower the drywell pressure, and thus, increases the DW-WW negative pressure. The small steamline break with DW spray operation is a new analysis which is potentially limiting for the negative containment pressure. The staff finds this change is acceptable because it was done to seek more conservative analysis for the containment negative pressure.

The results of the revised calculation show a significantly smaller calculated peak DW-WW negative differential pressure relative to the value reported previously, -3.86 versus -9.8 kilopascal (kPaD). GEH attributes this change to a less conservative analysis approach as described above. Based on its review the staff finds that the applicant's negative pressure design evaluation is consistent with SRP Section 6.2.1.1.C guidance, and therefore, is acceptable.

The results of the revised calculation show a smaller calculated peak wetwell-to-reactor building (WW-RB) negative differential pressure relative to the value reported previously, -8.76 versus -9.8 kPaD. The applicant attributes this to the SHEX code used to generate transient responses; the previous analyses used a series of end-point calculations to generate a set of conditions that produces a bounding prediction of the peak negative WW-RB differential pressure. The GEH states the following on using the SHEX code for calculating the WW-RB negative differential pressure:

> The GEH SHEX computer code was used for the revised calculations of the ABWR negative containment pressure for ABWR DCD Revision 5. The SHEX code has models for all containment, safety and auxiliary systems needed for the ABWR DCD negative pressure analysis. This is the code that corresponds to the Long-Term Cooling model identified in DCD Tier 2, Section 6.2.1.1.3.4.2. The GEH SHEX code has been verified and validated for general use in compliance with the GEH Nuclear Energy Quality Assurance Program.
>
> The GEH calculations of the ABWR containment negative pressure response with the SHEX code and evidence of verification for the calculations are contained within the GEH electronic archives of the design records.

Although, the original ABWR DCD did not name the computer code used for analyzing the containment long-term cooling, as stated above, GEH identified it as SHEX. Considering that the SHEX code has been verified and validated for general use, and it was used for analyzing the long-term containment response in the original ABWR DCD, Revision 4, the staff finds it acceptable to use the SHEX code for calculating the peak negative WW-RB differential pressure, which is another application of containment long-term response. As such, the staff finds GEH's response to Part (7) of RAI 06.02.01.01.C-1 acceptable.

The applicant provided the necessary information in the ABWR DCD, Revision 7, which incorporated the appropriate changes described in the applicant's response to Part (7) of RAI 06.02.01.01.C-1, that was found acceptable to the staff. Therefore, Confirmatory Item 6.2.3.1-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *6.2.1.3.4 Conclusion*

Based on the evaluation provided in this supplemental FSER section, the staff concludes that the changes in DCD Tier 2, Sections 6.2.1 and 6.2.2, related to short-term containment pressure response do not alter the safety findings made in NUREG–1503 and are consistent with SRP Section 6.2.1.1.C, Revision 7, issued March 2007. Therefore, the staff finds that the changes reviewed in ABWR DCD, Revision 7, resulting from containment re-analysis are acceptable and meet the requirements in GDC 16 and 50 and therefore are acceptable.

### *6.2.1.6 Suppression Pool Dynamic Loads*

#### *6.2.1.6.1 Regulatory Criteria*

The applicant added Combined License (COL) Information Item 3.8.6.5, "Loads Associated with Post-DBA [Design Basis Accident] Suppression Pool Water Level," to the DCD Tier 2, Section 3.8.6, "COL License Information." Because the applicant's change clarifies information in the original ABWR DC, it is a "modification," as this term is defined in Chapter 1 of this supplement. Therefore, this design change must comply with the regulations applicable and in effect at the time the certification was originally issued. The applicant's design change was made to correct an assumption on suppression pool water level used in hydrodynamic analysis.

The applicable regulatory requirement for evaluating the DCD change is based on 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 4, "Environmental and Dynamic Effects Design Bases (1997)," as it relates to the environmental and missile protection design. As pertinent here, GDC-4 requires that structures, systems, and components 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 and may occur during normal plant operation or following a loss-of-coolant accident (LOCA).

The staff reviewed the change using the guidance in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments," Revision 6, issued August 1984.

*6.2.1.6.2 Summary of Technical Information*

The applicant added the following to the GEH DCD Tier 2, Section 3.8.6, “COL License Information”:

> 3.8.6.5 Loads Associated with Post-DBA Suppression Pool Water Level: The COL applicants will confirm that the suppression pool water level used in the containment loads evaluation is based on the maximum predicted post-accident suppression pool water level rise that can occur concurrent with each of the defined containment loads (Appendix 3B). This load will then be used to update the associated analyses in Section 3.8, Appendix 3G and Appendix 3H.

*6.2.1.6.3 Regulatory Criteria*

In a letter dated March 31, 2014 (ADAMS Accession No. ML14090A068), the applicant provided the NRC a 10 CFR § 21.21(a)(2) “60-day interim report notification: “Containment Loads Potentially Exceed Limits with High Suppression Pool Water Level in the ABWR Design.” In Attachment 1 to the letter, the applicant stated the following:

> Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply[:] ABWR hydrodynamic loads have been calculated with the Suppression Pool water level defined at the Technical Specification Suppression Pool High Water Level (HWL). The Suppression Pool level during the postulated LOCA vessel blowdown may be greater than the Suppression Pool HWL during the pertinent timeframe for hydrodynamic loads because vessel coolant inventory is transferred into the suppression pool during blowdown. Additionally, certain containment structures previously thought uncovered may be submerged with the higher Suppression Pool water level. Increased hydrodynamic loads may correspondingly increase the totals in the design load combinations for which containment structures are designed to withstand.

In a letter dated August 29, 2014 (ADAMS Accession No. ML14241A306), the applicant informed the NRC that “[t]he GEH assessment has concluded that the predicted increase in the suppression pool water level above the value used for defining the ABWR loads and applied in the structural analysis will not result in the creation of a substantial safety hazard, nor will it lead to exceeding a technical specification [TS] safety limit for the US ABWR Certified Design.”

To determine the effect of this error on the GEH ABWR DC renewal application, in RAI 06.02.01-1 dated April 20, 2015 (ADAMS Accession No. ML15110A122), the staff requested that the applicant describe the impact of the error on loads on suppression pool wall boundaries, the access tunnel, and structures submerged in the suppression pool in terms of loads from pool swell, condensation oscillation, chugging, and safety relief valve (SRV) discharge.

GEH responded in a letter dated May 29, 2015 (ADAMS Accession No. ML15149A232), that the ABWR DCD, Appendix 3B, only identifies methods to be used in defining loads on submerged structures by citation to references. This includes the methods for loads due to LOCA pool swell, condensation-oscillation, chugging and SRV discharge.

According to the applicant:

(1) The loads affecting structural integrity, that are affected by the predicted increase in suppression pool water level are condensation oscillation and chugging;

(2) Pool swell loads are unaffected as they occur at the beginning of a LOCA before a significant transfer of water to the suppression pool that would raise the water level;

(3) The increase in pool boundary loads from SRV discharge due to the higher suppression pool water level is insignificant because the expected water level rise during SRV discharge is small; and

(4) The effect on SRV load is negligible relative to the conservatism in the SRV loads definition.

On October 28, 2015 (ADAMS Accession No. ML15357A292), the staff audited the applicant's analyses that was used as the basis for the DCD COL Information Item. The staff, in the course of the audit, determined that the applicant had identified the containment structural loads impacted by the predicted increase in suppression pool water level and has provided appropriate clarification for a COL applicant to perform the correct hydrodynamic load calculations that are based on a bounding predicted suppression pool water level.

The applicant evaluated predicted increases for LOCA condensation oscillation and chugging loads acting on the ABWR suppression pool boundaries. The applicant's evaluation along with the DCD COL Information Item will conservatively ensure, that a COL applicant will confirm the suppression pool water level used in the containment loads evaluation, based on the maximum predicted post-accident suppression pool water level rise. The applicant conservatively assumed that the predicted maximum suppression pool water level increase will result in increasing condensation oscillation and chugging forces by 50 percent and 20 percent. The resulting stresses in the reinforced concrete containment vessel and reactor pressure vessel pedestal for the governing faulted load combination will increase by less than 1 percent. The applicant concluded that potential increases or changes to hydrodynamic loads that were defined for the ABWR containment, that are associated with an elevated suppression pool water level, do not result in exceeding the original ABWR DC structural design limits.

The staff, as part of the audit, evaluated the applicant's analyses supporting this conclusion. The staff confirmed that the increase in resultant forces (less than 1 percent) due to the change in the level of the suppression pool water induced by the postulated LOCA event has a negligible effect on the containment structure loading.

On access tunnel structural integrity, the applicant's response in the letter dated May 29, 2015, states, "The access tunnel design is only described in the US ABWR DCD; there is no associated stress analysis results included in the US ABWR DCD."

On the integrity of submerged primary structure safety-related structures, components and equipment (SC&E) the applicant's May 29, 2015, response also stated the following:

> Increases in the [condensation oscillation] and chugging contribution to the emergency and faulted load combinations can result in increases in the primary structure model responses that can impact the design margins for safety-related

SC&E. The US ABWR DCD does not include design details for SC&E; there is no associated stress analysis results included in the US ABWR DCD.

Based on the applicant's response that the ABWR DCD does not contain the necessary design details of the access tunnel and submerged safety-related SC&E, the staff finds that the COL Information Item would provide the information for the detailed design to evaluate its impact.

In its response to RAI 06.02.01-1, the applicant stated that an evaluation of the access tunnel structural integrity was performed, for a non-domestic ABWR plant-specific design, in order to confirm that the predicted increase in the condensation oscillation and chugging loads do not result in exceeding the safety design margins of the access tunnel. The applicant stated that the evaluation determined that sufficient margins exist in the design to accommodate stress limits and buckling limits of the access tunnel.

As stated above in this SER Section 6.2.1.6.2, the applicant added COL Information Item 3.8.6.5 to ABWR DCD Tier 2, so that the COL applicant will use the appropriate suppression pool water level for the containment loads evaluation. The staff found this acceptable because the COL Information item directs the COL applicant to use the appropriate suppression pool water level for the containment load evaluation. However, the staff's acceptance of the ABWR design was not based on this COL Information Item; the existing DCD information is acceptable and revising the containment load evaluation, as confirmed in the staff audit, has negligible impact on the ABWR certified design.

Based on the review of the applicant's letter dated May 29, 2015, and the October 28, 2015, audit, the staff determined that the increased pool level induced by the postulated LOCA event does not have a significant impact on the design capacity of the containment structure and COL Information Item 3.8.6.5 will direct COL applicants to use the maximum predicted post-accident suppression pool water level rise that can occur concurrent with each of the defined containment loads in the designs of access tunnel and submerged primary structure safety-related SC&E. The staff concluded that the applicant addressed the staff's concerns raised in RAI 06.02.01-1, and therefore, the issue is closed.

The staff concluded that the containment structure, access tunnel, and primary structure safety-related SC&E meet the requirements of GDC 4 (1997).

#### *6.2.1.6.4 Conclusion*

The staff's review finds that the applicant's change to the ABWR DCD, Revision 7, is acceptable because it does not alter the safety findings made in NUREG–1503 and meets the applicable regulations in effect at the initial certification, including the requirements of 10 CFR Part 50, Appendix A, GDC 4.

### *6.2.1.9 Containment Debris Protection for ECCS Strainers*

#### *6.2.1.9.1 Regulatory Criteria*

In the GEH ABWR DCD, Revision 7, the applicant proposed a design change to the emergency core cooling system (ECCS) pump suction debris strainers. This supplemental evaluation documents the staff's review of the change to the ECCS strainer design described in DCD Tier 1, Tables 2.4.1, 2.4.2, and 2.4.4, and DCD Tier 2, Section 6C, "Containment Debris Protection for ECCS Strainers."

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 09 of the letter, the staff asked the applicant to confirm that the ECCS suction strainer design complies with 10 CFR § 50.46(b)(5), which included providing the net positive suction head (NPSH) margins determined using Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, issued March 2012, addressing chemical, in-vessel, and ex-vessel downstream effects, providing a structural analysis, and updating the inspections, tests, analyses, and acceptance criteria (ITAAC) as necessary consistent with the new guidance. In a letter dated September 17, 2012 (ADAMS Accession No. ML12261A311), the applicant informed the staff that it would address all the items identified in the July 20, 2012, letter.

The proposed changes do not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), this design change is an "amendment," as this term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations in effect at renewal. The applicable regulatory requirements for evaluating the design amendment to the ECCS strainers are given below.

The acceptance criteria for the performance of the ECCS following a loss-of-coolant accident (LOCA) are specified in 10 CFR § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The acceptance criterion dealing with the long-term core cooling phase of the accident recovery is 10 CFR § 50.46(b)(5), which states that:

> After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

As discussed in 10 CFR § 50.46(a)(1)(i), the ECCS must be designed so that the calculated cooling performance in the event of a LOCA resulting from a break in the primary reactor coolant system is in accordance with an acceptable evaluation model, or alternately, a model in conformance with the features of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." The primary ECCS safety functions are comprehensively modeled and evaluated for breaks up to and including the double-ended severance of a reactor coolant pipe to show that the ECCS will limit the peak clad temperature to below 1204 degrees Celsius (°C) (2,200 degrees Fahrenheit (°F)) and ensure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

The regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 1, "Quality Standards and Records," and 10 CFR § 50.55a, "Codes and Standards," require that systems and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. Regulations in 10 CFR § 50.55a also incorporate by reference the applicable editions and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, which addresses pressure integrity of components. Application of 10 CFR § 50.55a and GDC

1 provides assurance that established standard practices of proven or demonstrated effectiveness are used to achieve a high likelihood that these safety functions will be performed and that the codes and standards applied are commensurate with the importance to safety of these functions.

- GDC 4, “Environmental and Dynamic Effects Design Basis,” requires that structures, systems, and components (including pumps, valves, and strainers) important to safety accommodate the effects of and be compatible with the dynamic effects and environmental conditions associated with postulated accidents.
- GDC 34, “Residual Heat Removal,” requires that a system to remove residual heat be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
- GDC 35, “Emergency Core Cooling,” requires that an ECCS be provided that is capable of transferring heat from the reactor core following a loss of reactor coolant, at a rate sufficient to ensure that the core remains in a coolable geometry and that the clad metal-water reaction is limited to negligible amounts.
- GDC 38, “Containment Heat Removal,” requires that a system to remove heat from the reactor containment be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

The staff used the following guidance to determine whether the design of systems and components meets the regulatory requirements given above:

- RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident,” Revision 4, issued March 2012 (ADAMS Accession No. ML111330278), as supplemented by the NRC-approved Boiling Water Reactor Owners’ Group Utility Resolution Guidance (URG), NEDO-32686-A, “Utility Resolution Guidance for ECCS Suction Strainer Blockage,” Volumes 1 through 4, Revision 0, issued October 1998 (ADAMS Accession Nos. ML092530482, ML092530500, ML092530505, and ML092530507), provide guidance for boiling-water reactor (BWR) debris evaluations.
- Safety Evaluation by the Office of Nuclear Reactor Regulation, for Topical Report (TR) WCAP-16406-P-A, Revision 1, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191,’ Pressurized Water Reactor Owners Group, Project No. 694,” issued December 2007 (ADAMS Accession No. ML073520295).
- RG 1.100, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” Revision 3, issued September 2009, which endorses ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” (ADAMS Accession No. ML091320468).
- NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP), Section 6.2.2, “Containment Heat Removal Systems,” Revision 5, issued March 2007 (ADAMS Accession No. ML070160661).

### 6.2.1.9.2 *Summary of Technical Information*

In the following requests for additional information (RAIs), the staff requested information on various aspects of the proposed amendment to the ABWR ECCS pump suction debris strainer design. The following is a chronological list of these staff requests and the GEH responses. Details of the requests and responses is addressed in the specific sections where the issues were evaluated in the technical evaluation Section of this supplemental FSER.

In RAI 06.03-1, dated March 10, 2015 (ADAMS Accession No. ML15068A227), the staff requested that GEH, in accordance with 10 CFR § 52.59(a), provide information showing that the ECCS suction strainer design complies with 10 CFR § 50.46(b)(5). The applicant responded by letters dated April 8 and July 17, 2015 (ADAMS Accession Nos. ML15098A484 and ML15198A332).

In RAI 06.03-2, dated December 15, 2015 (ADAMS Accession No. ML15343A408), the staff requested that GEH provide detailed information in three areas (design and analysis of ECCS strainers, chemical effects, and downstream effects) showing that the ECCS suction strainer design complies with 10 CFR § 50.46(b)(5). The applicant responded by letters dated, May 27 and December 19, 2016, and February 23, 2017 (ADAMS Accession Nos. ML16148A101, ML16358A445, and ML17055C495 respectively).

On May 19, 2016 (ADAMS Accession No. ML16144A784), the staff had a public teleconference meeting with GEH regarding RAI 06.03-2 to provide clarification on the specific staff requests, which resulted in the GEH response on May 27, 2016.

On January 5, 2017, the staff had a public teleconference meeting with GEH regarding RAI 06.03-2 to provide clarification on the specific staff requests, which resulted in the GEH response on February 23, 2017.

In RAI 06.02.02-1, dated May 10, 2017 (ADAMS Accession No. ML17130A798), the staff requested the description of construction codes and classifications (safety class, ASME code class, seismic category and quality group) for the ECCS strainer design. The applicant responded in a letter dated June 16, 2017 (ADAMS Accession No. ML17167A161) and provided markups of the ABWR DCD, Revision 6, to the staff.

Following a public teleconference meeting with the staff on March 1, 2018 (ADAMS Accession No. ML18157A215), GEH provided an updated technical report (TR) NEDE-33878P, "ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability," Revision 3, issued March 2018, by a letter dated March 28, 2018 (ADAMS Accession No. ML18092A303).

In RAI 06.03-3 a follow up to RAIs 06.03-2 B1, B2, and B3, dated March 28, 2017 (ADAMS Accession No. ML17087A290), the staff requested additional information regarding the containment material and chemical affects. GEH responded to RAI 06.03-3 in a letter dated April 25, 2017 (ADAMS Accession No. ML17116A071) and provided markups to the ABWR DCD, Revision 6, in its response dated August 23, 2017 (ADAMS Accession No. ML17236A062), to address the staff follow up questions to RAI 06.03-2.

In RAIs 06.03-4 through 9, dated July 10, 2017 (ADAMS Accession No. ML17187A127), the staff requested that GEH provide additional information regarding the settling velocity of potential debris and system operation for these conditions. GEH responded by a letter dated August 23, 2017, with a revised response to RAIs 06.03-3 and RAIs 06.03-4 through 9.

The final DCD changes as a result of all RAIs included changes to DCD Tier 1, Tables 2.4.1, 2.4.2, and 2.4.4 and changed inspections, tests, and analysis design commitments for NPSH available at residual heat removal (RHR) system pumps, high pressure core flooder (HPCF) system pumps, and reactor core isolation cooling (RCIC) system pumps from "50% minimum blockage of the pump suction strainers" to "analytically derived values for blockage of pump suction strainers based upon the as-built system."

In addition, GEH provided the changes to DCD Tier 2, Chapter 6, Appendix 6C in markups to the ABWR DCD, Revision 6, as follows:

- Replaced ABWR ECCS suction strainers from using a "'T' arrangement with conical strainers on the two free legs of the 'T'" to a General Electric (GE) optimized stacked disk design in accordance with GEH licensing TR NEDC-32721P-A, "Application Methodology for the General Electric Stacked Disk ECCS Suction Strainer," Revision 2, issued March 2003 (ADAMS Accession Nos. ML031010390 (proprietary version) and NEDO-32721-A Revision 2 ML031010388 (public version)). This strainer design was utilized in response to NRC Bulletin (BL) 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," issued May, 1996, as a replacement of existing ECCS strainers with a large capacity passive strainer design. This design uses disks whose internal radius and thickness vary over the height of the strainer.
- Added evaluations of chemical effects and downstream effects, which were not considered during the original ABWR certification as these had not been discovered.
- Replaced RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 1, issued November 1985 (ADAMS Accession No. ML003740236), for sizing ECCS suction strainers to RG 1.82, Revision 4.
- Replaced DCD Tier 2, Table 6C-1, providing input parameters used for debris analysis with a table of ECCS strainer debris loads and deleted DCD Tier 2, Table 6C-2 providing ECCS strainers screen area and characteristic dimension.

#### *6.2.1.9.3 Technical Evaluation*

In FSER Section 6.2.1.9 of NUREG–1503, the staff discussed containment debris protection for the ECCS strainers (ADAMS Accession No. ML080670560). The FSER states that events at operating reactors involving the clogging of ECCS strainers led the staff to conclude that the guidance in RG 1.82, Revision 1, may not be conservative enough to eliminate this concern. The FSER mentions the issuance of Information Notice (IN) 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"; dated September 30, 1992; IN 93-34, "Potential for Loss of Emergency Cooling Function due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993; Supplement 1 to IN 93-34; dated May 6, 1993; and BL 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993. The staff indicated in the FSER that it was still working on resolving this issue for operating reactors. The staff stated that the issue regarding clogging of the ECCS strainers was resolved for the ABWR based on commitments made in Amendment 35 to the ABWR DCD. These commitments include:

- Sizing the RHR system suction strainers three times the area derived from NRC guidance for all breaks to account for uncertainty in the synergetic effects of strainer clogging from insulation, corrosion products, and other debris;
- Sizing the HPCF and RCIC system suction strainers according to guidance, but with conservatism in the mass of debris assumed to be deposited on the strainers; and
- Providing a 10-percent margin in the NPSH available from the static head of the suppression pool for conservatism.

However, because of lessons learned from BWR operating experience and during the review of Generic Safety Issue (GSI)-191, “Assessment of [Effect of] Debris Accumulation on PWR Sump Performance,” the staff determined that further review of the ECCS pump suction debris strainer design in the GEH ABWR DC renewal application was necessary to ensure continued compliance with the long-term cooling requirement of 10 CFR § 50.46(b)(5). The RG generally used for the ECCS debris strainer design was RG 1.82, Revision 0, issued June 1974, which included a 50-percent debris blockage criterion as a means to establish sufficient NPSH margin for the ECCS pumps. This criterion allows only 50-percent of the ECCS suction debris strainer to be clogged by debris. The certified design relies on an ITAAC that verifies this criterion. The later revisions of RG 1.82 provided guidance for designing an ECCS that includes the use of mechanistically determined debris head loss across the suction strainers.

NRC BL 96-03, asked BWR licensees to address potential debris plugging of ECCS suction strainers that were designed to meet the 50-percent debris blockage criterion. In response, licensees with operating BWRs replaced ECCS pump suction strainers with large-capacity passive strainers. The ABWR DC did not address BL 96-03 because of the timing of BL issuance and completion of the original ABWR DC review.

Efforts to address GSI-191 have led the NRC staff and the industry to identify new issues, including the effects of chemical precipitates impacting the performance of ECCS suction strainers, and downstream effects on fuel impacting the ability of cooling the reactor core following a LOCA.

**ECCS Suction Strainer Sizing Evaluation**

In RAI 06.03-2, Part A, the staff asked the applicant to provide design and analysis information for the ECCS strainer because it was missing in the ABWR DC renewal application. The applicant’s February 23, 2017, RAI response included a proprietary version of TR NEDO-33878, “ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability,” Revision 0 (ADAMS Accession Nos. ML17055C497 (public version) and ML17055C500 (proprietary version)), providing supporting technical information to show conformance with RG 1.82, Revision 4. The staff evaluation of the February 23, 2017, response to RAI 06.03-2 Part A, is given below. The staff’s evaluation of the remaining parts of the response is given under separate headings in this section.

In RAI 06.03-2, Part A.1, the staff asked the applicant to provide its evaluation of ECCS strainer performance (e.g., head loss) and provide the results of any analysis and/or tests performed in support of its findings. In response, the applicant stated that the ABWR ECCS strainers will be the patented GE optimized stacked disk design in accordance with NEDC-32721P-A Revision 2. Along with NEDC-32721P-A Revision 2, the applicant used an updated strainer debris head loss correlation to address an issue identified in a letter to the NRC, dated March 24, 2008 (ADAMS Accession No. ML080850242). The applicant proposed to update the ABWR DCD, "to remove obsolete information related to the T-shaped conical strainer, and outdated information such as the guidance to design for 50% plugging." The applicant provided the markup for DCD Tier 1, Tables 2.4.1, 2.4.2, and 2.4.4 replacing the 50 percent plugging criteria for NPSH available for RHR, HPCF, and RCIC pumps in the original ABWR certification to, "analytically derived values for blockage of pump suction strainers based upon the as-built system." The staff evaluated the markup and finds that proposed DCD Tier 1 changes are appropriate and are consistent with guidance in RG-1.82, Revision 4, and are therefore acceptable.

The staff focused on the applicant's use of the methodology in TR NEDC-32721P-A, Revision 2, which was previously approved by the staff for BWR ECCS suction strainer design and developed using an updated head loss correlation. The staff reviewed the updated head loss correlation as provided in NEDO-33878, Revision 3 (ADAMS Accession Nos. ML18092A306 (public version) and NEDE-33878P ML18092A308 (proprietary version)) and audited its derivation as given in reference documents listed in NEDO-33878. The staff determined that similarities between BWR Mark II and Mark III and ABWR containment designs would warrant using NEDC-32721P-A methodology for debris generation and transport. Quantities and types of debris causing the ABWR ECCS strainer head loss will be bounded by those for BWRs, except for chemical precipitates which have not been fully evaluated for BWRs. Chemical effects for the ABWR are addressed below under separate heading in this section. As such the NEDC-32721P-A, Revision 2 methodology is also applicable for the ABWR ECCS strainer head loss evaluation.

The staff reviewed the supporting technical information for strainer performance presented in NEDO-33878, Revision 3 to show conformance with RG 1.82, Revision 4 and performed a regulatory audit from February 21, 2017 – June 20, 2017, on other supporting documents as discussed in a regulatory audit summary report dated January 24, 2019 (ADAMS Accession No. ML18354B167), to confirm that the applicant has used the NEDC-32721P-A, Revision 2 methodology. The applicant also proposed to update DCD Tier 2, Appendix 6C summarizing analysis performed for the ECCS debris strainer and provided the associated ABWR DCD, Revision 6, markup. Based on the staff review of the applicant's response supported by the staff regulatory audit the staff determined that the applicant's response to RAI 06.03-2, Part A.1, was acceptable.

In RAI 06.03-2, Part A.2, the staff asked the applicant to provide the types and quantities of insulation debris being transported to the ECCS suction strainers and to the core following a design basis accident. This information is needed for evaluating the ECCS and core design heat transfer capabilities. In response, the applicant listed in a table, types and quantities of debris, determined in accordance with staff approved URG NEDO-32686-A, Revision 0, Volumes 1 through 4, and provided DCD Tier 2, markups in Table 6C-1. The staff review found that for sludge/corrosion products, inorganic zinc (IOZ), epoxy coated IOZ, rust flakes, and dust/dirt debris, the applicant has used quantities as recommended in Sections 3.3.2.2.2.1.1 and 3.2.2.2.3 of the URG and therefore, the listed debris types and quantities specified for the ABWR are acceptable.

During the regulatory audit summarized in the January 24, 2019, audit report, the staff confirmed that the applicant used reflective metallic insulation and Nukon fiber insulation quantities calculated for a reference ABWR plant using a methodology recommended in the URG. Therefore, considering that the applicant has a design commitment to use "analytically derived values for blockage of pump suction strainers based upon the as-built system" for ECCS pumps, the staff determined that the applicant's response to RAI 06.03-2, Part A.2, was acceptable.

In RAI 06.03-2, Part A.3, the staff asked the applicant to provide details of the ECCS debris strainer for assessing its performance under accident conditions because this information was not provided in the ABWR DC renewal application. In response, the applicant stated that the conical strainer design used in the original ABWR DC was obsolete and was updated to the GEH stacked disk strainer and design details related to the stacked disk strainer performance and sizing methodology can be found in the TR NEDC-32721P-A, Revision 2, which applies an updated head loss correlation. The applicant provided the following ECCS suction strainer configuration used for the ABWR application:

- Type: GEH stacked disk passive suction strainer
- Flow Area: Each strainer has perforated area 36 $m^2$ (388 $ft^2$) with 20 disks [combined surface area of 216 $m^2$ (2328 $ft^2$) for three (3) RHR, two (2) HPCF and one (1) RCIC strainer]
- Hole Size: 3.2 mm (0.125 inch) diameter.

As described in the January 24, 2019, regulatory audit report, the staff reviewed the proposed new strainer drawing to confirm the above information and the supporting sizing calculations. The staff determined that the applicant's response to RAI 06.03-2, Part A.3, is acceptable because it provided requested information in conformance with RG 1.82, Revision 4.

In response to RAI 06.03-1, GEH in its letter dated April 8, 2015 (ADAMS Accession No. ML15098A487), proposed to delete DCD Tier 2, Tables 6C-1 and 6C-2, which provided debris analysis input parameters and results of ECCS debris strainer sizing analysis without providing alternate tables or references to calculation reports. The staff needed the referenced information provided in those tables in the ABWR DCD to support its review. Therefore, in RAI 06.03-2, Part A.4, the staff asked the applicant to provide the corresponding information. In its response letter dated May 27, 2016, the applicant proposed to add DCD Tier 2, Table 6C-1 providing design basis debris load (i.e., types and quantities of debris) used in sizing ECCS strainers. The applicant stated that the DCD Tier 2, Table 6C-1 information, combined with the methodology in NEDC 32721P-A, Revision 2, provides the necessary inputs to design a strainer that complies with 10 CFR § 50.46(b)(5). The staff reviewed the applicant's response and supporting documentation during the staff regulatory audit and determined that the applicant has used the NEDC-32721P-A, Revision 2, methodology in conformance with RG 1.82, Revision 4, and therefore, the applicant's response to RAI 06.03-2, Part A.4, was acceptable.

In its response to RAI 06.03-1, dated July 17, 2015, the applicant provided references to guidance documents, for example, "Of the debris generated, the amount that is transported to the suppression pool shall be determined in accordance with [NEDO-32686-A] based on similarity of the Mark III upper drywell design." However, the response did not provide the ECCS debris strainer design input calculated using these guidance documents nor did it reference calculation reports providing such information. Therefore, in RAI 06.03-2, Part A.5, the staff asked the applicant to provide an analysis documenting the implementation of this guidance.

In the applicant's response to RAI 06.03-2, Part A.1, dated February 23, 2017, GEH provided a markup to DCD Tier 2, Appendix 6C adding a reference to NEDC-32721P-A, Revision 2 (with a note explaining the updated head loss correlation), which provides the strainer design methodology. As evaluated above the staff determined that the applicant provided information on analysis and implementation of NEDC-32721P-A, Revision 2 consistent with RG 1.82, Revision 4, and therefore, the applicant's response to RAI 06.03-2, Part A.5, is acceptable.

Based on the review of the applicant's submittal, including RAI responses and the staff regulatory audit summary, the staff finds that the GEH ABWR ECCS suction strainer sizing evaluation conforms to the guidance in RG 1.82, Revision 4, NEDO-32686-A, Revision 0, and TR NEDC-32721P-A, Revision 2, and meets the requirements in 10 CFR § 50.46(b)(5), and is, therefore, acceptable.

The applicant provided the necessary information from RAI 06.03-2, Part A.1, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response. Therefore, Confirmatory Item 6.2.1.9-1, from the staff advanced safety evaluation report (SER) with no open items for the ABWR DC renewal, is resolved and closed.

**ECCS Suction Strainer Structural Evaluation**

The staff reviewed TR NEDC-32721P-A, Revision 2, for conformance to RG 1.82, Revision 4. In this TR, the applicant addressed hydraulic performance design methods and provided procedures for the calculation of hydraulic loads for new strainer installations. Hydrodynamic loads in the suppression pool are directly caused by the movement of suppression pool water, driven by the oscillation of air and/or steam bubbles at either the locations of the main LOCA vents or the safety relief valve (SRV) discharge. The applicant states that the bubble source closest to the strainer will be the dominant source of hydrodynamic loads to the strainers. In all cases, a location scale factor is calculated based on the nearest bubble source. The scaled loads (resulting from multiplying a location scale factor to the load created from collapsing bubbles) are then multiplied by dynamic load factors (DLF) that are calculated from the natural frequencies of the new strainers. The new DLFs are based on the frequency ratio between the frequency of the bubble source and the natural frequency of the strainer assembly. The DLFs for suddenly applied loads (SRV Jet, LOCA Jet, and Fallback) are taken at 2.0. The product of the scale factors, DLFs, and the original loads is taken to be the load on the new strainer. The staff finds the methodology of assessing the hydrodynamic loads of the new strainers as provided in NEDC-32721P-A, Revision 2, meets the guidance of RG 1.82, Revision 4, and is therefore acceptable.

The staff reviewed design specifications and design documents of ECCS strainers during the staff's regulatory audit, to verify that the component design meets the methodology and criteria described in DCD Tier 2, Section 6.2.2, and that the design of ECCS strainers conforms to RG 1.82, Revision 4, and the design requirements have been properly translated to the design documents. The structural analysis of the strainer is performed to ensure the structural adequacy of the strainer, and includes seismic, differential pressure, and hydrodynamic loads. This set of loading categories is consistent with those applied to ASME BPV Code Class 1, 2, and 3 components as described in DCD Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and is therefore acceptable to the staff. During the regulatory audit, the staff found that design specifications and design documents of ECCS strainers are consistent with the methodology and criteria described in the DCD Tier 2, Section 3.9.3, for ASME Code Class components and component supports. During the regulatory audit of the ABWR ECCS strainer design, the staff noticed that, in GEH Document 24A5822, "ECCS Suction Strainers, Piping and Support," Revision 7, July 21, 1999 *(GEH Proprietary*), the applicant specified the ECCS strainers are designed in accordance to ASME Code Section III. However, DCD Tier 2, Section 6C, does not provide the information of construction codes and standards of the ECCS strainers design and Table 3.2-1, "Classification Summary," did not include the component classifications of ECCS strainers (e.g., safety class, ASME code class, seismic category and quality group). In RAI 06.02.02-1, dated May 10, 2017 (ADAMS Accession No. ML17130A798), the staff requested the applicant provide in DCD Tier 2, the description of construction codes and classifications (safety class, ASME code class, seismic category and quality group) for the ECCS strainer design.

In the RAI response dated June 16, 2017 (ADAMS Accession No. ML17167A161), the applicant states that it added the component classification for the ABWR ECCS suction to DCD Tier 2, Table 3.2-1, as provided in the ABWR DCD, Revision 6, markups. In the markups to DCD Tier 2, Table 3.2-1, the ECCS pump suction strainers in the ABWR suppression pool are classified to be Safety Class 2, Location C, Quality Group Classification B, Quality Assurance Requirement B, seismic Category I. The applicant added DCD Tier 2, Table 3.2-1, with the following notes (ii):

> ASME BPV Code Section III, Class 2 requirements are used as guidance for specification development of the design, fabrication, and inspection of the ECCS pump suction strainers, commensurate with the safety importance of the strainers. The strainers are not required to be ASME Code stamped and no ASME Certificate of Authorization is required (the strainers do not function as a pressure boundary and are attached to the end of the piping within the suppression pool). In addition, if required, the strainers may be supported from the suppression pool wall and floor.

The staff finds that the markup to DCD Tier 2, Table 3.2-1, is acceptable, since the component classification of the strainer design is appropriate to ensure the design, fabrication, erection, construction, testing, and inspection for the strainer will be commensurate with the importance of the safety function to be performed which is in accordance with GDC 1. Therefore, RAI 06.2.2-1 is closed.

The applicant provided the necessary information in the ABWR DCD Revision 7 which incorporated the changes described in the applicant's letter dated June 16, 2017, to address RAI 06.2.2-1. Therefore, Confirmatory Item 6.2.1.9-2 from the staff's advanced

SER with no open items for the ABWR DC renewal is resolved and closed. Based on the reviews of TR NEDC-32721P-A, Revision 2, on the proposed amended ABWR strainer design and the staff regulatory audit of ECCS strainer design, the staff finds that GEH ABWR ECCS suction strainer design includes the appropriate loads and is compatible with the environmental conditions associated with normal, operation, maintenance, testing, and postulated accident loads, including LOCAs and therefore meets GDC 4 requirements and conforms to the guidance in RG 1.82, Revision 4, and NEDC-32721P-A, Revision 2.

**Chemical Effects Evaluation**

*Chemical Effects Introduction and/Background:*

The term "chemical effects" refers to the possibility that interactions between materials and the post-LOCA containment environment will generate chemical precipitates that may contribute to blockage and head loss at the strainers and/or reactor core. For pressurized-water reactors (PWRs), the staff published detailed guidance in 2008 for evaluating plant-specific chemical effects (ADAMS Accession No. ML080380214). This includes guidance on using WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," issued March 2008 (ADAMS Accession No. ML081150383). Conforming to this guidance provides one acceptable way to meet the requirements of 10 CFR § 50.46, as they relate to the effect of chemical precipitates on the ECCS for PWRs.

The NRC has not issued comparable chemical effects guidance to BWR licensees or applicants. The generation of chemical precipitates in the water chemistry representative of a BWR post-LOCA environment has not been thoroughly studied. In RG 1.82, Revision 4, the staff Regulatory Position 3.3.1, states that post-LOCA containment conditions for BWRs may result in chemical interactions different than those considered for operating PWRs. The BWR Owners Group (BWROG) has performed testing to begin evaluating the chemical effects issues with operating BWRs.

The BWROG performed benchtop testing as an initial step to quantify material release rates, which refers to the release of elements into solution through corrosion of metallic materials and dissolution of non-metals. The BWROG report, "Review of Boiling Water Reactor Material Dissolution in Post-LOCA Containment Systems," issued November 2013 (ADAMS Accession No. ML14328A639), describes the results of the BWR testing. The NRC has not reviewed that testing to determine how it applies to new or operating reactors. To evaluate chemical effects for the ABWR DC renewal, the staff reviewed the information in the application and supplemental information in letters dated July 17, 2015, May 27, 2016, December 19, 2016, February 23, 2017, April 25, 2017, and August 23, 2017.

The complexity of evaluating chemical effects for BWRs is increased by the uncertainty in the post-LOCA chemical environment, which may be pH-buffered in the alkaline range or unbuffered, depending on the licensing basis of the plant and the conditions during the accident. Alkaline buffering is used by some plants, to prevent post-LOCA iodine re-evolution, and it is accomplished with addition of the borated chemical solution in the standby liquid control system (SLCS). Iodine re-evolution at acidic pH conditions is a potential consequence of acids generated by radiolysis of water, air, and organic materials (e.g., cable jacketing). However, even without an intentional buffer addition, the pH may become more alkaline if fiberglass is destroyed during the LOCA and subsequently undergoes chemical dissolution.

One possible difference for BWRs is that the lower pH range of an unbuffered post-LOCA fluid compared to the alkaline-buffered PWR fluid could lead to more corrosion of zinc (e.g., galvanized steel) and carbon and low-alloy steels. The ABWR containment has zinc in the form of galvanized steel and IOZ coatings, and some uncoated steel piping in the ECCS. It is not known whether zinc and iron form precipitates that clog fiber beds, other than the steel corrosion products ("sludge") that are included as part of the debris load in the resolution of BWR ECCS clogging in NEDO-32686, Revision 0, in response to NRC BL 96-03. The material called "sludge" is composed of iron oxide corrosion products in the suppression pool, and it was included as a type of debris in NEDO-32686, Revision 0, based on the observation of this material in operating plants and its role in the clogging events at Barseback Unit 2 (Sweden, 1992), Perry Nuclear Power Plant (1993), and Limerick Generating Station Unit 1 (1995). It may originate from exposed steel surfaces in the suppression pool or from connected piping systems.

*Applicant's Approach to Addressing Chemical Effects*

The applicant described its overall approach to chemical effects in DCD Tier 2, Section 6C.3.2, which states that the ABWR is designed to preclude the materials and environmental conditions most likely to generate chemical precipitates that may contribute to blockage and head loss. It also states that aluminum, phosphate, and calcium silicate will not be in containment. Both statements are based on the applicant's letter dated December 19, 2016, with corresponding ABWR DCD, Revision 6, markups and reflected in ABWR DCD, Revision 7.

The applicant considered IOZ and epoxy coatings in the particulate debris load (DCD Tier 2, Table 6C-1) but not for chemical effects. Other materials present in the containment that could potentially contribute to chemical effects are fiberglass, concrete, steel, and zinc as galvanized steel. The applicant did not identify any chemical effects associated exclusively with the post-LOCA environment. Therefore, sludge was the only corrosion product material assumed to be circulating in the post-LOCA fluid that potentially contributes to clogging of fiber beds.

The applicant described its treatment of carbon and low-alloy steel corrosion products in its letter dated August 23, 2017. The revised response to RAI 06.03-2 states that wetted surfaces are stainless steel or stainless-clad steel. The ABWR also uses stainless steel pipe in some connected systems. The design limits the surface area of bare carbon steel using protective coatings. The sludge is attributed to the corrosion of carbon steel piping and components in the ECCS during normal operation. For sludge, the design value of a net accumulation of sludge inside containment of 100 pounds mass (lbm) per year (45.4 kilograms (kg) per year during normal operation is between the median (88 lbm/39.9 kg) and mean (129 lbm/58.5 kg) values among BWRs surveyed for the URG. The applicant notes that assuming a net accumulation of 100 lbm sludge per year inside containment should be considered reasonable based on limiting the amount of carbon and low-alloy steel and the fact that the suppression pool clean-up system (SPCS) removes particulates and dissolved impurities to very low levels. Combined license (COL) Information Item 6.2.7.3, directs the COL applicant to address methods for maintaining the level of cleanliness assumed in the strainer debris evaluation.

The design of the GEH ABWR is based on 200 lbm (90.8 kg) of sludge, as listed in DCD Tier 2, Table 6C-1, which represents a two-year operating cycle. In the letter dated August 23, 2017, the applicant described the basis for assuming 200 lbm for the ABWR. The URG stated that each licensee should assume an initial sludge generation rate to be used in evaluating the ECCS suction strainers, since the actual rate requires measurements over a period of time. The URG suggested an assumed value of 150 lbm per year, about 1.7 times the median rate measured in the 1995 survey of BWR licensees. The applicant assumed 200 lbm for a two-year cycle as a reasonable assumption. It is less than the amount recommended in the URG but near the mean (258 lbm) and median (176 lbm). The ABWR has features designed to reduce the amount of sludge generation compared to older plants. For example, it includes a stainless steel lined suppression pool, less use of carbon steel piping connected to the suppression pool, and a SPCS that maintains water quality equivalent to the Fuel Pool Cooling and Cleanup System (e.g., less than 30 parts per billion corrosion product metals, less than 1.2 micro Sieverts (μS)/centimeter (cm) conductivity (a measure of the impurity level), and pH 5.6 – 8.6). In addition, COL Information Item 6.2.7.3, requires COL applicants to address acceptable methods for maintaining the level of cleanliness assumed in the ECCS strainer debris evaluation. According to DCD Tier 2, Section 6.2.1.7, the cleanliness methods will include removing, at periodic intervals, debris that might not be removed by the SPCS.

The applicant described the basis for including no chemical effects from concrete in its letter dated December 19, 2016. In response to RAI 06.03-2 B.1.e, the applicant stated that all concrete is coated and protected from jet impingement by a liner plate.

According to DCD Tier 2, Section 6.1, the ABWR design uses mostly metal-reflective thermal insulation in containment, but it includes 23.4 kg (51.6 lbm) of fiberglass insulation. This quantity is stated to be small relative to the amount reported in a survey of operating plants in 2015 (from information in Chapter 6 of the FSER for the COL application for South Texas Project, Units 3 and 4 dated September 29, 2015 (ADAMS Accession No. ML120830102)), which ranged from roughly 9 to 1600 kg (20 to 3600 lbm). In addition, the fiberglass is used only on small diameter piping and dispersed; therefore, no single break would affect all of it. The ABWR design was also identified by the applicatnt to exclude other types of non-metallic insulation found to contribute to strainer head loss and chemical effects in PWRs, such as calcium silicate and microporous insulation.

The applicant described its approach to zinc from galvanized steel in its letter dated August 23, 2017. The letter provided details of a corrosion calculation for galvanized steel surfaces in containment. The calculation includes estimated values of the surface area that would be wetted in a post-LOCA environment and the corresponding corrosion rate. The applicant concluded that the zinc released over the 30-day period would remain dissolved in the post-LOCA environment rather than form a precipitate.

The applicant described the pH and temperature conditions in its letter dated August 23, 2017, in response to RAI 06.03-2 B.1. The response states that the pH range will be maintained between 5.3 and 8.9 based on DCD Tier 2, Section 3I.3.2.3, "Water Quality and Submergence," which lists the pH and other reactor water quality characteristics for design basis LOCAs. It explains that the contents of the SLCS, although intended for beyond design basis accidents, could be added during the post-LOCA period to prevent pH below the design range. Such use of the SLCS requires operating procedures that would be developed by a COL applicant according to DCD Revision 6, Tier 2, Section 13.5.

The letter dated August 23, 2017, also described the suppression pool temperature. In response to RAI 06.03-2 B.2, the applicant stated that the temperature may increase to 77°C (170°F) at 30 minutes and may later reach a maximum of 89°C (192°F). The response refers to DCD Tier 2, Section 6.2.1.1.3.3, and Figures 6.2-7 and 6.2-15. Figure 6.2-15 extends to 28 hours, at which time the suppression pool temperature is decreasing and about 70°C (158°F).

*Staff Evaluation of Chemical Effects*

In its letter dated August 23, 2017, and in other responses, the applicant described maintaining this design basis pH range (5.3 – 8.9) as a "flat time history." Based on the corrosion rate variation for some ECCS materials over this range, the staff evaluated the applicant's proposed pH range and does not consider it to be a flat pH profile. For example, in the PWR chemical effects methodology, WCAP-16530-NP-A, the release (corrosion) rate of aluminum increases about five-fold from pH 5.3 to 8.9 at 82°C (180°F). Therefore, the evaluation below discusses the effect of the pH range where appropriate. The staff notes that for carbon steel and zinc, corrosion rates decrease as the pH increases within the range 5.3 to 8.9, assuming other factors are constant.

The applicant identified sludge as the only corrosion product material circulating in the post-LOCA fluid. Sludge has been included as a debris source for BWRs since the issue of ECCS strainer clogging was first identified. Sludge has not been identified as a "chemical effect" for BWRs, since the focus of chemical effects for PWRs has been on post-LOCA chemical reactions and because the industry and the NRC staff have reached no conclusions about BWR chemical effects. The staff evaluated 200 lbm (91 kg) of sludge as chemical debris in the ABWR standard design based on the following:

- ECCS strainer clogging events have been attributed to sludge combined with fibrous insulation.
- Characterization of sludge indicates it has properties similar to chemical precipitates studied for PWRs.
- Blockage of flow through fiber beds in laboratory testing has been attributed to steel corrosion products.
- BWROG strainer testing with a bed of fiberglass insulation and simulated sludge produced a sustained pressure drop. The staff's SER for the URG described the role of sludge in testing and in events at BWRs at certain operating conditions when the ECCS was in service. The "Background" section of the SER has the following observations and conclusions about strainer clogging:
    - Barseback Unit 2 (Sweden, 1992) – a pipe break can generate and transport insulation and other debris to the ECCS strainers and cause loss of NPSH.
    - Perry (1993) – fibrous debris combined with corrosion products in the suppression pool (sludge) can exacerbate the loss of NPSH.
    - Limerick Unit 1 (1995) – A diver found suction strainers covered with a thin mat of material consisting mostly of fibers and sludge (iron oxides).

- Alden Research Laboratory – testing to support understanding of these BWR strainer events confirmed that fibrous debris filtering sludge greatly increases pressure drop across the ECCS strainer.

Studies of ECCS clogging included detailed characterization of BWR sludge. Results of these studies are documented in NUREG/CR-6367, "Experimental Study of Head Loss and Filtration for LOCA Debris," issued February 1996, and include the following:

- BWR suppression pool sludge as more than 99 weight percent steel corrosion particulate material. Some larger particles were postulated to result from agglomeration of an amorphous gelatinous component.
- The approximate sludge particle size based on characterization performed by the BWROG: 81 weight percent 0–5 micrometers (μm), 14 weight percent 5-10 μm, and 5 weight percent 10-75 μm. The smallest particles were approximately 0.1 μm diameter. (1 μm = $4 \times 10^{-5}$ inch).
- The study proposed two mechanisms for blockage in the fiber beds-based on photomicrographs of the beds: smooth coating of fibers with small particles and blockage of passages by agglomerates.

The staff evaluated these results and they indicate that sludge has effects on fiber beds similar to the chemical effects recognized in testing in PWR environments. More recent studies suggest iron corrosion products formed in the post-LOCA environment (as opposed to the operating environment) can cause pressure drops in fiber beds. Testing performed by Framatome in Germany with galvanized steel in flowing, acidic boron-containing (boric acid) solutions at $50^{0}$C (122°F) resulted in clogging of a mineral wool filtering bed with corrosion products of both zinc and steel (H. Ludwig and F. Roth, "Influence of Corrosion Processes on the Protected Sump Intake after Coolant Loss Accidents," Nuclear Technology Annual Convention 2006, English translation (ADAMS Accession No. ML083510156).

The staff also noted that for the tests with the liquid streaming onto the galvanized surface, the results indicated that the galvanized coating was physically removed, and the resulting steel corrosion products accumulated in the mineral wool caused a pressure drop. The significance of this testing for the ABWR is the suggestion that iron corrosion, which occurs under acidic conditions, may contribute to clogging of a fiber bed. Any specific effect of boron in these results would probably not be applicable to the ABWR since boron would only be present in the ABWR post-LOCA fluid as a result of adding sodium pentaborate from the SLCS. The sodium pentaborate would produce a mildly alkaline pH that inhibits iron corrosion.

In Japan, the Nuclear Electric Safety Organization (JNES) sponsored chemical effects testing, mostly in PWR environments but with one test in a BWR environment (see Section 4.3.1.3 of "Fiscal 2007 PWR Sump Screen Chemical Effect Test," Japan Nuclear Energy Safety Organization, issued May 2008 (ADAMS Accession No. ML090410358). The test in the BWR environment, with a pH range between 3.2 and 6.5, produced a high concentration of iron in the test solution from corrosion of the carbon steel. When passed through a fiber bed, the iron-rich test solution produced a significant pressure drop. The pH range in the JNES test overlaps the design basis range (5.3 – 8.9) for the ABWR, and the pressure drop decreased as the pH increased from 3.2 into the ABWR range.

Pressure drops from precipitated iron, as well as from zinc and aluminum, in a fibrous bed were also observed in vertical loop tests sponsored by the NRC (NUREG/CR-6868, "Small-Scale Experiments: Effects of Chemical Reactions on Debris-Bed Head Loss," issued March 2005 (ADAMS Accession No. ML050900260). The tests were conducted in boron-containing solutions at about 25 to 45$^{0}$C (77 to 113°F) and a room-temperature pH of approximately 7.

Based on the operating events, testing observations, and characteristics listed above, the staff finds that sludge can be considered a chemical effect for the ABWR design by the way it is formed by the reaction of containment materials and environment and can cause head loss by clogging flow paths in fiber beds. Sludge is different than the PWR chemical effects in the staff-approved PWR methodology in that it can be formed during both operation and post-LOCA. As noted above, characterization found that about 95 percent of sludge particles are less than 10 μm diameter. By comparison, LOCA-generated particulate debris and latent debris are expected to be mostly larger than 10 μm ($4 \times 10^{-4}$ inch) based on tests and sampling. The small size of sludge particles, combined with the conclusion in NUREG/CR-6367 (Appendix B) that sludge particles can both coat fibers and agglomerate to block larger gaps, makes sludge behavior similar to that of PWR chemical precipitates.

The design of the GEH ABWR is based on 200 lbm (90.8 kg) of sludge, which is listed in DCD Table 6C-1, and represents a two-year operating cycle. In its letter dated August 23, 2017, the applicant describes the basis for assuming 200 lbm for the ABWR. The URG stated that each licensee should assume an initial sludge generation rate to be used in evaluating the ECCS suction strainers, since the actual rate requires measurements over a period of time. The URG suggested an assumed value of 150 lbm per year, more than 1.5 times the mean rate measured in the survey of the BWR licensees in 1995. The applicant assumed 200 lbm for a two-year cycle as a reasonable assumption. It is less than the amount recommended in the URG but near the mean (258 lbm) and median (176 lbm).

The ABWR has features designed to reduce the amount of sludge generation compared to older plants. For example, it includes a stainless steel lined suppression pool, less use of carbon steel piping connected to the suppression pool, and a SPCS that maintains water quality equivalent to the Fuel Pool Cooling and Cleanup System (e.g., less than 30 parts per billion corrosion product metals, less than 1.2 μS/cm conductivity (a measure of the impurity level), and pH 5.6 – 8.6). In addition, COL Information Item 6.2.7.3, directs COL applicants to address acceptable methods for maintaining the level of cleanliness assumed in the ECCS strainer debris evaluation. According to DCD Tier 2, Section 6.2.1.7, the cleanliness methods will include removing, at periodic intervals, debris that might not be removed by the SPCS.

During a LOCA, the ABWR SPCS would be isolated as part of containment isolation. Therefore, iron corrosion products formed on the surface of the carbon steel ECCS piping and components could potentially contribute to head loss if a fiber bed forms on the strainers and fuel inlet. Because of the design features for limiting accumulation of iron corrosion products (sludge) during operation, the staff considers it reasonable to expect there is margin in the 200 lbm of sludge to account for corrosion of carbon steel and low-alloy steel following a LOCA. COL Information Item 6.2.7.2 directs the COL applicant to provide confirmation that the 200 lbm limit can be achieved in the as-built plant. In its review of the COL application for South Texas Project, Units 3 and 4, the staff audited operating experience from ABWRs in Japan, and concluded that 200 lbm was a conservative assumption for the sludge quantity. This was documented in the corresponding STP 3 & 4 FSER Section 6.2.1.4 (ADAMS Accession No. ML120830102).

To address zinc, the applicant in its letter dated August 23, 2017, provided a detailed calculation of zinc corrosion from galvanized steel, and the corresponding zinc concentration in the post-LOCA fluid for 30 days. The applicant determined a zinc release (corrosion) rate was determined using the results of laboratory tests performed in demineralized water by the BWROG (R. W. Eaker and S. G. Sawochka, BWROG Report NWT 863, "Review of Boiling Water Reactor Material Dissolution in Post-LOCA Containment Solutions," Revision 0, NWT Corporation, November 2013 (ADAMS Accession No. ML14328A635). The applicant used a zinc release rate of 0.05 grams per square meter per hour, which is high relative to most of the measured values for test temperatures between 140 and 200$^0$F (60 and 93$^0$C). To estimate the galvanized surface area that would be exposed to the post-LOCA fluid, the applicant started with the highest value of galvanized steel in containment reported by an operating plant in an operating BWR survey [NRC Public Meeting Slides, "BWROG ECCS Suction Strainers Committee," December 2, 2015 (ADAMS Accession No. ML15335A419)]. By multiplying this surface area by the estimated fraction of the area that would be wetted and the zinc release rate, the applicant calculated a total of 3.4 pounds of zinc released. Based on published values of zinc solubility [R. A. Reichle, K. G. McCurdy, and L. G. Helper, "Zinc Hydroxide: Solubility Product and Hydroxy-complex Stability Constants from 12.5-75 °C," Canadian Journal of Chemistry, Vol. 53 (1975), pp. 3841-3845.], the applicant concluded all of the released zinc would remain in solution and not form a precipitate (3.4 lbm released compared to at least 4.8 lbm solubility) (1.5 kg compared to at least 2.2 kg).

At the time of publication of this supplemental FSER, the staff has not evaluated the BWROG testing and release rates to determine if the results are realistic or conservative. Higher corrosion rates are listed for solid zinc samples in aerated distilled water over the same temperature range [D. C. H. Nevison, "Corrosion of Zinc," in Metals Handbook, 9th Edition, Vol. 13 (ASM International, Metals Park, Ohio, 1987), pp. 759-761]. In addition, no determination has yet been made by the BWROG, individual licensees, or the NRC staff, that zinc corrosion leads to a precipitate that causes strainer head loss. This level of uncertainty prevents the staff from concluding that released zinc would remain dissolved in the post-LOCA fluid and not contribute to chemical effects. However, because the applicant assumes a large quantity of sludge that is known to cause strainer head loss, which would provide margin in the event of zinc precipitation, and because it is not known if zinc precipitates cause head loss, the staff finds it acceptable to neglect incremental chemical effects from zinc corrosion for renewal of the ABWR DC.

The staff finds it acceptable to neglect chemical effects from concrete for the ABWR based on all concrete being either protected with a liner plate or coated and outside the zone of influence for the coating. In the suppression pool, the liner that separates the concrete from the water is stainless steel. In addition, the staff notes that the amount of chemical precipitate from concrete, calculated using the PWR chemical effects methodology for the ABWR temperatures over the pH range 5.3 – 8.9, is small (of the order of 0.01 kg per 100 square meters ($m^2$)) (0.02 lbm per 1,000 square feet ($ft^2$)). Based on the protection of concrete surfaces in the design and the small contribution to chemical effects, the staff concluded that it is acceptable for the ABWR to assume any chemical effects from concrete are negligible with respect to the overall quantity of chemical reaction products (i.e., 200 lbm sludge).

The ABWR design also includes 23.4 kg (51.6 lbm) of fiberglass insulation. As with zinc, for BWRs it is not known if fiberglass in the post-LOCA fluid contributes to chemical precipitates and strainer head loss. The BWROG chemical effects testing [R. W. Eaker and S. G. Sawochka, BWROG Report NWT 863, Revision 0], measured releases of silicon, calcium, sodium, and aluminum from fiberglass in demineralized water at BWR temperatures. This report was submitted for information, so the staff has used it for insights but has not formally endorsed it. The WCAP-16530-NP-A methodology for PWRs predicts chemical precipitate from fiberglass insulation, and the amount of precipitate from 23.4 kg of fiberglass is less than one kilogram (or 2.2 lbm) of sodium aluminum silicate at pH 5.3 to a few kilograms (or few pounds) at pH 8.9, using an estimated temperature profile. It is not yet known whether fiberglass insulation produces chemical effects for BWRs, but given the amount predicted for PWRs, the staff finds it acceptable for the applicant to neglect incremental chemical effects from fiberglass. The BWROG testing also showed fiberglass increased the pH of test solutions in the BWROG testing. A pH increase would be beneficial in terms of steel and zinc corrosion, although the amount of fiberglass insulation in the ABWR design may be too small to raise the pH significantly.

Based on the discussion above for the materials included in the design and those excluded, the staff concluded that 200 lbm (91 kg) of sludge is acceptable for the ABWR renewal. Since only iron corrosion products have been considered as chemical precipitates causing strainer head loss and reactor core pressure drop, this approach may not be suitable for other BWR designs and licensees. For example, BWRs with a combination of high pH and aluminum can be expected based on the PWR research to produce chemical precipitates, but the industry and the NRC staff have not yet determined how to conservatively quantify and evaluate them under BWR conditions.

*Chemical Effects Evaluation Summary*

Unlike for PWRs, chemical effects for BWRs have not been fully evaluated and defined in terms of staff-approved industry guidance. For PWRs, there is a methodology for quantifying and evaluating chemical effects in terms of $AlOOH$, $Na_2AlSi_3O_8$, and $Ca_3(PO_4)_2$. For BWRs, which have a lower pH without SLCS addition, the BWROG has studied zinc and iron corrosion products. The corrosion of galvanized steel and bare carbon or low-alloy steel can be higher or lower at unbuffered BWR conditions than at PWR conditions, depending on pH and temperature. The industry has not determined if this corrosion generates chemical reaction products that cause head loss. For the ABWR renewal, GEH included only iron corrosion product sludge in the ECCS suction strainer qualification. The staff considered the properties of sludge and operating experience with sludge causing clogging of suction strainers when combined with a fiber bed. The staff evaluated the applicant’s sludge quantity and the basis for neglecting additional chemical effects and, based on the evaluation above, found it acceptable. In addition, COL Information Item 6.2.7.3 directs a future COL applicant to address acceptable methods for maintaining the level of cleanliness assumed in the ECCS strainer debris evaluation.

The applicant provided the necessary information from RAI 06.03-2 in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant’s response related to DCD Tier 2, Section 6C.3.2. Therefore, Confirmatory Item 6.2.1.9-3 from the staff advanced SER with no open items for the ABWR DC renewal is resolved and closed.

**Ex-vessel Downstream Effects Evaluation**

The term “ex-vessel downstream effects” refers to effects of post-LOCA debris on the systems and components in the ECCS flow path (excluding the reactor vessel) located downstream of the ECCS strainers in the suppression pool. By letters dated August 23, 2017 and March 28, 2018 (ADAMS Accession No. ML17236A060 and ML18092A293 respectively), GEH provided markups to ABWR DCD Tier 2, Revision 6, Section 6C.3.3, “Downstream Effects,” and incorporated by reference NEDE-33878 Revision 3, to evaluate the impact of post-LOCA debris on the ABWR components downstream of the ECCS strainers. Areas of concern addressed for ex-vessel downstream effects include: (1) blockage of system flow paths at narrow flow passages (e.g., ECCS sparger spray nozzles, pump internal flow passages, and tight-clearance valves), and (2) wear and abrasion of surfaces (e.g., pump running surfaces) and heat exchanger tubes and orifices.

The NRC staff reviewed the applicant’s evaluation of ex-vessel downstream effects for conformance to RG 1.82, Revision 4, to provide reasonable assurance that the ECCS components will function as designed under post-LOCA fluid conditions for the required mission time. The staff guidance in Section C.1.1 of RG 1.82, Revision 4, specifies regulatory positions common to all water-cooled reactors and Subsection C.1.1.10 states that the NRC considers the staff approved SER for WCAP-16406-P, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191,” dated December 20, 2007 (ADAMS Accession No. ML073520295), evaluation methods and criteria to be acceptable for components downstream of the sump strainers. Therefore, NRC staff applied the methodologies for the evaluation of the downstream ex-vessel components as approved in the staff SER for WCAP-16406-P to the GEH ABWR reactor design. The following sections of this FSER supplement, provide the staff’s evaluation.

*ECCS Systems and Components*

NEDE-33878P, Revision 3, Section A.2, “ECCS System Descriptions and Mission Times,” describes the ECCS systems in the scope of the downstream ex-vessel effects evaluation. The ABWR ECCS consists of the HPCF, the steam driven RCIC, and the RHR systems that take suction through the ECCS strainers. NEDE-33878P, Revision 3, Table A-1, “ECCS Mission Time and Description,” identifies the ECCS long-term and short-term system operating lineups, conditions of operation, and mission times. NEDE-33878P, Revision 3, Tables A-4 through A8 identify and evaluate the effects of post-LOCA debris on the systems and components in the scope of the downstream ex-vessel effects evaluation. The NRC staff evaluated the revisions and found that the GEH NEDE-33878P, Revision 3, Section A.2, descriptions of mission time, systems and components is consistent with the guidance of RG 1.82, Revision 4 and, therefore, acceptable.

*Post-LOCA Fluid Constituents*

NEDE-33878P, Revision 3, Appendix A.3, “Debris Ingestion,” and Table A-2, “ABWR Debris Source Term,” describe the type, size, and quantity of debris that is small enough to pass through the holes of the ECCS suction strainer perforated plates.  The ECCS suction strainer hole size is 0.125 inch.  NEDE-33878P, Revision 3, Table A-3, “ABWR Debris Downstream Concentration,” describes the debris concentration in the ECCS fluid.  The NRC staff evaluated the revisions and determined that the type, size, and quantity of debris assumed to bypass the sump strainer is consistent with RG 1.82, Revision 4, and the staff SER for WCAP-16406-P and is therefore acceptable.

*RHR, HPCF, and RCIC Pump Evaluation*

ABWR DCD Tier 2, Section 3.9.6.1 specifies the design conditions under which pumps will be required to function but does not specifically address post-LOCA debris conditions under which pumps will be required to function.  Therefore, the staff issued RAI 06.03-6 on July 10, 2017, requesting that GEH address design and qualification requirements for the pumps during post-LOCA operation in DCD Tier 2, Section 3.9.6.1. In its response to RAI 06.03-6, (ADAMS Accession No. ML17236A062) GEH provided a markup for revisions to DCD Tier 2, Section 3.9.6.1, to specify qualification of the ECCS pumps (including mechanical seals) under all design basis conditions including post-LOCA conditions is validated under ASME QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” as endorsed by RG 1.100, Revision 3, “Seismic Qualification Of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” issued September 2009.

In addition, GEH provided a markup for DCD Tier 1, Table 2.4.1, ITAAC 4.c, for the RHR pumps; Table 2.4.2, ITAAC 3.g for the HPCF pumps; and Table 2.4.4, ITAAC 3.j for the RCIC pump, to specify that the test result/report must confirm that the pumps perform their intended function during post-LOCA operation.  The post-LOCA debris conditions in the ECCS fluid are specified in NEDE-33878P, Revision 3.  The staff finds the GEH response acceptable because ASME Standard QME-1-2007 as endorsed in RG 1.100, Revision 3, provides an acceptable methodology for the functional qualification of pumps for post-LOCA operation and the ITAAC specify that the test result/report confirm that the pumps perform their intended function during post-LOCA operation.  The staff determined that the pump evaluation confirms that the testing is adequate to ensure the pumps meet the regulatory requirements of GDC 4, to be compatible with the environmental conditions associated with LOCAs.

The applicant provided the necessary information from RAI 06.03-6 in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant’s response. Therefore, Confirmatory Item 6.2.1.9-4 from the staff advanced SER with no open items for the ABWR DC renewal is resolved and closed.

*Heat Exchanger Evaluation*

NEDE-33878P, Appendix A, Tables A-4 through A8, Revision 1, issued May 2017 (ADAMS Accession Nos. ML17132A029 (proprietary version) and NEDO-33878 ML17132A028 (public version)), describe the effect of post-LOCA debris on the operation of the RHR heat exchangers.  The table column titled, “Auxiliary Equipment Evaluation,” state that flow from the suppression pool is channeled through the shell side of the RHR heat exchangers and concludes that the heat exchangers will operate as designed during post-LOCA operation.  However, GEH did not address the effects of post-LOCA debris on the shell side of the RHR heat exchanger.  Therefore, in RAI

06.03-7, dated July 10, 2017, the staff requested that GEH address the effects of post-LOCA debris on the shell side of the RHR heat exchanger. In its response to RAI 06.03-7, GEH stated that ABWR DCD Tier 2, Section 5.4.7.1, describes a design change that the ABWR RHR heat exchanger has reactor water flowing through the tube side of the heat exchanger. The primary purpose of the change was to reduce radiation buildup in the heat exchanger by providing a more open geometry flow path through the center of the tubes, as opposed to the shell side construction of spacers, baffles, and low flow velocity locations, which can provide places for radioactive sludge to accumulate. In the RAI response, GEH described that debris size, debris characteristics, and the flow velocities through the heat exchanger will preclude plugging, fouling, wear, and debris settling. GEH also stated that the RHR heat exchanger specifications require the vendor to meet performance requirements under design debris loading conditions that will be validated through the procurement process with a certificate of compliance. GEH revised the TR NEDE-33878P, Revision 2, issued August 2017 (ADAMS Accession Nos. ML17236A064 (proprietary version) and ML17236A063 (public version)), to clarify the reactor water (debris) flow path through the heat exchanger tubes.

The staff evaluated and determined that the GEH methodology to evaluate RHR heat exchanger plugging, fouling, wear, debris settling, and heat transfer performance in the presence of post-LOCA debris is acceptable because the effect of debris on the heat exchanger is consistent with the methodologies approved by the staff in the SER for TR WCAP-16406-P and the vendor will provide a certificate of compliance to verify conformance to performance requirements.

The staff confirmed that the applicant provided the necessary information from RAI 06.03-7, in the ABWR DCD, Revision 7, which incorporated the appropriate changes described in the applicant's response. Therefore, the staff determined that RAI 06.03-7, is closed and resolved.

*Blockage Evaluation for Components Such as Valves, Orifices, Pipes, and Spray Nozzles/Spargers*

NEDE-33878P, Revision 2, Appendix A, Tables A-4 through A8 describe the evaluation for blockage of valves, orifices, spray nozzles/spargers, and pipes during operation with post-LOCA fluids. GEH states that the ECCS piping and component flow area exceeds the maximum dimension of the debris particles and that blockage is not expected for valves, orifices, pipes, and spray nozzles/spargers during operation with post-LOCA fluids. However, GEH did not address the potential blockage for tight-clearance valves that may not be in the fully open position during post-LOCA operations. Therefore, the staff in its RAI 06.03-9, dated July 10, 2017, requested that GEH address the potential blockage for tight-clearance valves. In its response to RAI 06.03-9, dated August 23, 2017, GEH stated that the RHR, HPCF and RCIC systems do not have any throttle valves that are susceptible to blockage because all throttle valves will be in the open or closed position. GEH also stated that the check valves are installed on the suction and discharge of the pumps and due to valve opening clearances during operation are not susceptible to blockage.

The staff evaluated and determined that the GEH methodology of evaluation for blockage of valves, orifices, spray nozzles/spargers, and pipes during operation with post-LOCA fluids is acceptable because the flow diameters are larger than the maximum debris size and the evaluation is consistent with the methodology approved in the SER for TR-WCAP-16406-P.

The staff confirmed that the applicant provided the necessary information from RAI 06.03-9, in the ABWR DCD, Revision 7, which incorporated the appropriate changes described in the applicant's response. Therefore, the staff determined that RAI 06.03-9, is closed and resolved.

*Instrument Tubing Blockage Evaluation*

NEDE-33878 Revision 1, Appendix A, Tables A-4 through A-8 describe debris settling in instrument lines during post-LOCA operation for the ABWR design. In the column titled "Fluid Velocity Through Component," GEH states it is assumed that settling (instrument sensing lines/components) will occur when the flow velocity is less than the settling velocity for the debris type. Therefore, the staff in its RAI 06.03-5, dated July 10, 2017, requested that GEH provide additional information to address any instrument lines where debris settling, and blockage may occur. In its response to RAI 06.03-5, dated August 23, 2017, GEH stated that it revised NEDE-33878P Revision 2, to clarify that the ECCS instrument lines in service during post-LOCA operation are installed above the horizontal plane of the process piping and that no settling of debris in the instrument tubing is expected in this configuration.

The staff evaluated and determined that the GEH evaluation for instrument tubing is acceptable because the ECCS instrument lines in service during post-LOCA operation are installed above the horizontal plane of the process piping and no settling or ingestion of debris in an instrument line with this configuration is expected. The instrument tubing evaluation is consistent with the methodologies approved by the staff in the SER for TR-WCAP-16406-P.

The staff confirmed that the applicant provided the necessary information from RAI 06.03-5, in the ABWR DCD, Revision 7, which incorporated the appropriate changes described in the applicant's response. Therefore, the staff determined that RAI 06.03-5, is closed and resolved.

*Wear Evaluation for Components Such as Pumps, Valves, Orifices, Pipes and Spray Nozzles/Spargers*

NEDE-33878, Revision 2, Appendix A, Tables A-4 through A-8, states that the effect of post-LOCA debris on component and system wear for the mission time is insignificant. However, GEH did not describe a methodology to determine that wear for individual components is acceptable during post-LOCA operation. Therefore, in its RAI 06.03-8, dated July 10, 2017, the staff requested that GEH describe the methodology to determine that wear for individual components is acceptable for post-LOCA operation.

In the applicant's response to RAI 06.03-8, dated August 23, 2017, GEH stated that experimental data on the effects of particulates applied to ECCS type pumps show that degradation in pump performance is negligible for particulate concentrations less than 1 percent by volume as referenced in NUREG/CR-2792, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," issued September 1982. NUREG/CR-2792 notes conservative estimates of the nature and quantities of debris show that fine abrasives may be present in the concentrations of about 0.1 percent by volume (about 400 parts per million by weight) and that very conservative estimates of fibrous material yield concentrations of less than 1 percent by volume. GEH stated that the debris concentrations specified in in NUREG/CR-2792 are less than the downstream debris concentrations during post-LOCA operation as specified in NEDE-33878P, Revision 2, Table A-3. The NRC staff evaluated the information provided and considers this response acceptable for the pumps because the wear methodology described by GEH indicates that pump internal component wear such as the impeller and bearings will not lead to pump performance degradation and that pump performance under all design basis conditions including post-LOCA debris loading conditions will be validated by qualification under ASME Standard QME-1-2007, as endorsed by RG 1.100, Revision 3.

The staff confirmed that the applicant provided the necessary information from RAI 06.03-8, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response. Therefore, the staff determined that RAI 06.03-8, is closed and resolved.

For non-rotating equipment such as piping, valves, heat exchangers, spargers, and instrumentation tubing, NEDE-33878P, Revision 3, Appendix A.1, states that wear and abrasion of surfaces in the ECCS during post-LOCA operation are evaluated based on the flow rates to which the surfaces are subjected and the grittiness or abrasiveness of the ingested debris. GEH concluded that the expected wear of non-rotating ECCS components such as piping, valves, heat exchangers, spargers and instrumentation tubing during the post-LOCA mission time under design basis debris loading will not adversely impact the ECCS performance. The NRC staff evaluated the information provided and considers this response acceptable for wear and abrasion of components in the ECCS flow path because the components are evaluated based on the abrasiveness of the debris and the ECCS flow rates and the expected wear will not impact ECCS performance. The staff, therefore, finds that the component wear evaluation is consistent with the methodologies approved by the staff in the SER for TR-WCAP-16406-P.

*Debris Settling Evaluation for Valves, Orifices, Pipes, and Spray Nozzles*

The staff SER for TR-WCAP-16406-P addresses debris settling and accumulation of debris in low flow areas that may occur when the settling velocity of the debris is less than the minimum flow velocity in the system piping and components. If the system/component flow velocity exceeds the debris settling velocity, it is assumed that the minimal settling of debris will occur and performance of the ECCS components will not be adversely impacted. Therefore, the staff in its RAI 06.03-4, dated July 10, 2017, requested that GEH provide additional information to identify any areas where settling velocity of the debris is less than the minimum flow velocity in system/components and provide the basis for acceptable system operation for these conditions. In the applicant's response to RAI 06.03-4, dated August 23, 2017, GEH stated that the flow through ECCS piping and components under design basis conditions exceeds the debris settling velocity with significant margin, therefore, debris settling is not expected to occur.

The staff evaluated and determined that the GEH response stating that flow through ECCS piping and components under design basis conditions exceeds the debris settling velocity with significant margin is acceptable because debris settling is not expected to occur when the system flow velocity exceeds the debris settling velocity and the evaluation is consistent with the methodologies approved by the staff in the SER for TR-WCAP-16406-P.

The staff confirmed that the applicant provided the necessary information from RAI 06.03-4, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response. Therefore, the staff determined that RAI 06.03-4 is closed and resolved.

*Chemical Effects Evaluation for Ex-Vessel Downstream Components*

The term chemical effects refer to the possibility that interactions of materials in the containment environment will generate chemical precipitate debris that may contribute to blockage in systems including downstream ex-vessel components. In RAI 06.03-2, dated December 15, 2015, the staff requested that GEH provide additional information regarding the chemical effects during post-LOCA operation. In its last revised response to RAI 06.03-2 dated February 23, 2017, GEH described the chemical effects evaluation and stated that the interaction of materials is not expected to generate chemical precipitation debris in the ABWR containment environment. GEH also stated that zinc chemical debris in very small quantities that could result from the corrosion of IOZ coating was assumed to transport to the suction strainer and that this debris is included as sludge in debris source term specified in NEDE-33878P, Revision 2, Table A-2.

The staff evaluated and determined that the applicant's evaluation that chemical precipitants have no effect on plugging or wear of downstream ex-vessel components is acceptable because it is consistent with staff positions documented in NRC memorandum, "Basis for Excluding Chemical Effects Phenomenon from WCAP-16406-P Ex-vessel Downstream Evaluations," issued January 21, 2010 (ADAMS Accession No. ML093160100), and the NRC TR, "Evaluation of Chemical Effects Phenomena Identification and Ranking Table Results," issued March 2011 (ADAMS Accession No. ML102280594).

The staff confirmed that the applicant provided the necessary information from RAI 06.03-2, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response. Therefore, the staff determined that RAI 06.03-2 is closed and resolved.

**Ex-Vessel Downstream Effects Evaluation Summary**

The NRC staff reviewed the provision in ABWR DCD and NEDE-33878P, Revision 3, for ex-vessel downstream effects and, verified the inclusion of the necessary information in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response to RAI 06.03-2. Therefore, Confirmatory Item 6.2.1.9-3 from the staff advanced SER with no open items for the ABWR DC renewal is resolved and closed.

The NRC staff concludes that the provisions for mitigating downstream effects meet the regulatory requirements in GDC-4 for the components downstream of the ECCS strainer to be compatible with the environmental conditions associated with LOCAs. This conclusion is based on the applicant having specified provisions in the ABWR DCD and NEDE-33878P, Revision 3, that the ECCS design meets the staff approved methodology in the SER for TR-WCAP-16406-P, RG 1.82, Revision 4, and ASME Standard QME-1-2007, which contain approved methodologies for satisfying the GDC 4 requirements.

**In-Vessel Downstream Effects Evaluation**

*Introduction/Background*

The evaluation of in-vessel downstream effects of debris on long-term post-LOCA core cooling includes consideration of potential blockage at the core inlet, either at the core support plate structure or at the inlet nozzle of individual fuel bundles, collection of debris on bundle grid spacers, buildup of fibrous, chemical, and protective coating debris on fuel rod cladding surfaces, blockage of in-core bypass flow paths, and the phenomena associated with the various phases of LOCA, including blowdown, reflood, and post-reflood. The ABWR reactor vessel internals and fuel design are similar to those of conventional BWRs, so the studies and tests performed by the BWROG to address the effects of debris on ECCS performance, and the lessons learned from them, are generally applicable to the ABWR design. Numerous ABWR design enhancements serve to minimize the effects of debris on fuel cooling following LOCAs. These are described by GEH in DCD Tier 2, Chapter 6, and are evaluated in this FSER supplement.

The NRC has not formalized review guidance for the evaluation of in-vessel effects of debris. At the time the agency issued RG 1.82, Revision 4, in 2012, the review of in-vessel downstream debris effects in PWRs was in progress. The PWR Owners Group had submitted for review TR WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid." As noted in RG 1.82, Revision 4, a method and reference for PWR licensees whose plants are bounded by its input assumptions to use in evaluating the downstream impact of debris on the performance of fuel following a LOCA, are subject to conditions and limitations specified in the NRC SER. The staff had not completed its review of WCAP-16793-NP when it issued of RG 1.82, Revision 4. Also, the applicability to BWRs had not yet been addressed, and no in-vessel testing with debris had been conducted for BWRs.

For BWRs, the staff has accepted the BWROG-sponsored URG for ECCS Suction Strainer Blockage, NEDO-32686-A. Volume 4 of that document contains a GEH generic SER that addresses the potential for fuel bundle flow blockage and consequent fuel damage. The staff SER for the URG (contained in Volume 1 of NEDO-32686-A) does not specifically address fuel blockage. Also, the URG does not specifically address ABWR ECCS design features, the certified ABWR fuel, or reactor vessel internals. However, since the ABWR fuel, core design, and ECCS are similar to those of currently operating conventional BWRs, the URG conclusions should generally be applicable. This is discussed further in the staff evaluation below.

*Applicant's Approach for Addressing In-vessel Downstream Effects*

In RAI 06.03-2, Part C, dated December 15, 2015, the staff requested that GEH justify the acceptability of the ABWR core design and certified fuel with respect to core cooling in the presence of debris, including test results and/or analysis to support the design.

GEH responded to this question on February 23, 2017, April 25, 2017, and August 23, 2017, with downstream effects evaluation described in Appendix A.5 of TR NEDE-33878P, Revision 0, submitted with the letter dated February 23, 2017. The applicant transmitted subsequent Revisions 1, and 2 to NEDE-33878P through separate letters April 25, 2017 and August 28, 2017. Following a public teleconference meeting with the staff on March 1, 2018 (ADAMS Accession No. ML18157A215), GEH provided an updated TR NEDE-33878P, Revision 3, by a letter dated March 28, 2018 (ADAMS Accession No. ML18092A303). The primary technical change between Revision 0, and subsequent revisions of the TR is the reduction of ECCS "mission time" from 100 days to 30 days following a LOCA. The staff typically considers 30 days to be an appropriate time period to demonstrate ECCS functionality, since beyond that time the decay heat loading will be small, such that alternative cooling will be possible should ECCS functionality be lost.

Appendix A.5, of GEH TR NEDE-33878P, Revision 3, discussed the in-vessel flow paths that could be blocked by debris following a LOCA. These include the normal and bypass flow paths between the reactor vessel lower plenum and upper plenum and through the fuel bundles. Table A-4 of the TR provides a qualitative evaluation of the effects of debris on reactor vessel internals and fuel, as well as on ECCS components. Since the minimum dimension of internal flow paths exceeds the largest debris particle dimension, GEH concludes that clogging is not considered credible. The TR references the URG, which qualitatively assesses fuel bundle inlet blockage. It is stated that if debris totally blocks the inlet to one or more fuel channels from below, these bundles would receive radiation cooling to the channel walls as the bypass refills, then direct cooling from water spill-over from above once the water level is restored above the top of the fuel channels.

NEDE-33878P, Revision 3, notes several ABWR design improvements from currently-operating BWRs that should minimize the effects of debris. The ABWR design eliminates the use of recirculation piping external to the reactor pressure vessel (typical of conventional BWRs) by use of reactor internal recirculation pumps. This reduces the likelihood of a large high-energy line break, which can be a significant source of debris for conventional BWRs, and, although not stated by GEH, eliminates a large coolant leakage path below the top of active fuel (TAF). In addition, main steam and feedwater piping connect to the RPV above the TAF. The possibility of a large break LOCA below the TAF is therefore eliminated. Also, the ABWR design has diverse ECCS delivery points, which helps reduce the consequences of downstream blockage. Two HPCF loops deliver coolant to the region above the core within the core shroud. One of three low pressure core flooder loops (LPFL) provides coolant through one of the feedwater lines. The RCIC system delivers coolant to the other feedwater line. Two LPFL systems deliver coolant through separate spargers into the outer annulus region.

As noted in NEDE-33878P, Revision 3, tests with a small concentration of fibrous insulation material were performed to assess the potential blockage of coolant flow at the entrance to fuel assemblies. Modern GEH nuclear fuel (GNF2) was used for the tests. The evaluation concluded that significant BWR fuel bundle inlet clogging does not result in GNF2 heat up after the LOCA refill from ECCS injection. GEH states that this conclusion applies to other BWR fuel bundles (such as the ABWR GEH GE P8x8R nuclear fuel) with an equivalent degree of inlet resistance.

*Staff Evaluation of In-vessel Downstream Effects*

To evaluate the GEH response to RAI 06.03-2 and the accompanying TR NEDE-33878P, Revision 3, it is necessary to have a detailed understanding of ABWR design features and the similarities and differences from conventional BWRs. It is also important to note key differences from PWRs, for which the majority of in-vessel testing, and analyses has been done. BWRs use channeled fuel assemblies (bundles) which inhibit cross-flow, while PWR core designs allow open channel flow between fuel assemblies. Each ABWR fuel bundle has an independent flow path between the lower and upper plenum of the reactor vessel. BWR and PWR ECCS designs differ significantly with respect to diversity of injection locations. Both PWR and BWR fuel utilize grid spacers to maintain the relative position of fuel rods but differ in the number and size of fuel rods and number of spacers.

In DCD Tier 2, Figure 5.3-2a, “Reactor Pressure Vessel Key Features,” shows the relative locations of vessel internal components and indicates the various ECCS injection locations via HPCF, LPFL, and feedwater spargers (used for RHR core cooling Mode A1 and RCIC). DCD Tier 2, Table 4.4-1 “Typical Thermal-Hydraulic Design Characteristics of the Reactor Core,” provides relevant thermal-hydraulic design characteristics of the reactor core, including coolant flow area per assembly, core average and maximum inlet velocity, and total core pressure drop. DCD Tier 2, Table 4.4-5, “Reactor Coolant System Geometric Data,” provides average flow areas for the upper and lower plenum, core, and downcomer.

The applicant described the ABWR reactor pressure vessel internals in DCD Tier 2, Section 3.9.5, “Reactor Pressure Vessel Internals,” and the mechanical and nuclear design of the fuel and reactor core are described in DCD Tier 2, Chapter 4. The high and low-pressure systems which comprise the ECCS are described in DCD Tier 2, Chapter 5, and DCD Tier 2, Section 6.3 “Emergency Core Cooling Systems.” The ABWR design includes many features that will minimize the sources of debris and the mechanisms for transport into the ECCS and subsequently to the fuel. These are described in various places in the DCD and are summarized below.

The ABWR core and fuel are described in DCD Tier 2, Chapter 4. The core is comprised of 872 channeled fuel assemblies surrounded by a cylindrical core shroud. The annular (downcomer) region between the core shroud and the inner reactor vessel wall serves as the primary coolant flow path, both during normal operation and following a LOCA. During normal operation, feedwater flow and recirculation flow are directed downward to the lower plenum of the reactor vessel and then upward through the core. Flow enters each fuel assembly through a side entry orifice in the core support assembly, and then is directed upward to each channel through a transition (nose) piece to the lower tie plate (LTP). Fuel rods are held in place by the LTP, and spacing is maintained throughout the length of the channel box by grid spacers distributed over the bundle length. An upper tie plate is used at the bundle exit. Flow within the channel box is axial along the fuel rods in the open area between them. Alternate core flow paths include gaps between fuel bundles and between peripheral bundles and the inner reactor vessel wall. Each fuel bundle also includes two small channel-to-LTP bypass holes. Design details for the certified ABWR P8x8R fuel are provided in TR NEDE-31152P, "General Electric Fuel Bundle Designs Evaluated with GESTAR-Mechanical Analysis Bases (proprietary version)," issued December 1988 and Supplement 1, issued June 2000 (ADAMS Accession Nos. ML003725063 and Supplement 1 ML003725068, (non-publicly available)). Design parameters which affect the fuel cooling and potential capture of debris include the fuel rod cladding outer diameter, number of rods and rod pitch, channel box dimensions, and number and type of grid spacers.

To confirm the GEH assertion that the clogging of reactor vessel internals and fuel with debris is not credible, the staff reviewed the limiting dimensions of internal flow paths and fuel flow areas and compared them to the physical dimensions of the various types of debris that could be transported to the reactor vessel, as specified in the URG (NEDO-32686-A) and NUREG/CR-6367. Since any debris that can bypass the suction strainer must be smaller than 0.125 inches in diameter to pass through the strainer, this dimension was used for the assessment. Although some local clearances are less than this width, the staff agrees with GEH that complete blockage of a fuel bundle is not credible. This is because there are multiple possible ways in which ECCS water can reach the fuel rods within each channel. Even if the inlet to a fuel bundle is completely blocked, ECCS water can replace any liquid mass lost due to boiloff by downflow of spray droplets or spillover from the top of the channel box from the upper plenum. The upper plenum above the core serves as a common mixing region for all bundles, so even if one or several fuel channels are blocked, cooling liquid can be supplied from adjacent unblocked bundles.

The past BWR fuel bundle head loss test results show that the highest debris pressure drop occurs at the location of the first or the second spacer location if the fluid flows from the fuel bundle bottom to the top. The debris bed gradually forms at these two locations after the injected ECCS water from the strainer flow through these two spacers and the downward edge of the spacers captures the debris. Initially, the debris only accumulates along the edge of the spacers with the flow area away from the spacers open to the fluid. With more and more debris gradually piling up on the debris already captured by the spacers, a porous debris bed could form to bridge the gap between the spacers and the fuel pins. The porosity of the debris bed affects the pressure drop across the bed and is strongly affected by the number of spacers along the fuel bundle width. The more spacers included in the fuel bundle design, the denser the debris bed is, and the higher the pressure drop expected.

The staff compared the GNF2 fuel bundle design tested by Global Nuclear Fuel (GNF) to the ABWR fuel bundle design referenced by the DC renewal application, which is designated as GE P8x8R fuel, and noted two main differences. The GNF2 fuel bundle design has more fuel pins and an additional spacer to capture debris. Also, the GNF2 fuel bundle design incorporates a debris filtering LTP. The staff evaluated both of those differences to determine whether it agrees that the GNF2 testing results are applicable to the ABWR P8x8R fuel design.

With respect to the number of fuel pins and spacers, since the tested fuel bundle had a larger number of fuel pins, the debris captured in this configuration would tend to be greater than for the ABWR P8x8R design with fewer debris capture locations. Therefore, should the ABWR P8x8R fuel be tested for the same given amount of debris and fluid flow rate, the pressure drop across the debris bed would be expected to be less, so adequate cooling could be maintained.

With respect to the LTP design, the tested GNF2 fuel bundle would tend to capture more debris than the P8x8R fuel design without a debris filtering LTP. However, as previously stated, only debris with the size of 0.125” or less can pass through the strainer to get into the core region, and regardless of the LTP design, the holes in the debris filter of both the GNF2 and the P8x8R fuel are large enough to pass the debris bypassing the strainer. The GNF2 would capture more debris than the ABWR P8x8R design. As it was observed in the GNF2 fuel bundle head loss testing, the total measured pressure drops remained less than the available driving head.

Considering both major applicable design differences between the tested GNF2 fuel bundle and the ABWR P8x8R fuel design, the staff concludes that the hydraulic performance of the ABWR P8x8R fuel bundle is bounded by the results of fuel bundle head loss testing for the GNF2 fuel. Therefore, the staff concludes that debris blockage at the bundle inlet and lower grid spacers would not adversely affect the P8x8R fuel used by the ABWR.

It has been common nuclear industry practice to evaluate the effect of fuel cladding thermal resistance post-LOCA caused by the buildup of layers of oxide, crud, and chemical precipitates and the possible occurrence of a second peak cladding temperature during long-term cooling. Cladding oxidation in the ABWR design following a LOCA will be insignificant since the core remains covered. The high turbulence resulting from boiloff of liquid in the core region is expected to impede significant buildup of solid particulate, fiber and chemical precipitates on cladding surfaces. Because the ECCS injects water from both the top and bottom of the core, it is unlikely that a significant quantity of any solid particulate, fiber and chemical precipitate would be deposited on the fuel cladding surface. Therefore, the staff considers that an increase in thermal resistance from oxide, crud and chemical precipitates will be minimal, and occurrence of a second peak cladding temperature during post-reflood long-term cooling is unlikely.

For the reasons stated above, the staff finds the applicant's approach for evaluation of in-vessel downstream effects for the ABWR design is sufficient to demonstrate that the long-term cooling requirement of 10 CFR § 50.46(b)(5) is satisfied when considering the effects of debris.

#### *6.2.1.9.4 Conclusion*

Based on the above evaluation, the staff concluded that the design of the GEH ABWR ECCS suction strainer meets all applicable regulations including GDC 1, 4, 34, 35, and 38. Additionally, based on the above, the staff concluded that the design as described in the ABWR DCD, Revision 7, conforms to the guidance in RG 1.82, Revision 4, and TR NEDO-32686-A, Revision 0 and TR NEDC-32721P-A, Revision 2. The staff determined that the GEH ABWR ECCS suction strainer design complies with 10 CFR § 50.46, specifically the requirements of 10 CFR § 50.46(b)(5) and is, therefore, acceptable.

## 6.3 Emergency Core Cooling Systems

### *6.3.1 Regulatory Criteria*

In the GEH ABWR DCD, Revision 6, the applicant had not changed the loss-of-coolant accident (LOCA) analysis to incorporate error corrections and other changes to the emergency core cooling system (ECCS) evaluation model (EM) that had been identified since the original ABWR DC in 1997. As discussed below, GEH has made changes to the DCD to account for these error corrections and other changes to the ECCS EM in ABWR DCD, Revision 7.

In a letter dated July 21, 2016 (ADAMS Accession No. ML16174A175), the NRC staff made GEH aware that reported ECCS EM changes and errors for the ABWR standard plant design had not been accounted for in Revision 6 of the ABWR DCD submitted in February 2016. Therefore, the staff requested that GEH provide DCD changes to meet the requirements of 10 CFR § 52.57(a), which specifies that the renewal application must contain all information necessary to bring up to date the information and data contained in the previous application. In a letter dated August 19, 2016 (ADAMS Accession No. ML19031C851), GEH committed to addressing the issue in Revision 7 of the DCD.

Because the applicant's changes correct errors in the ABWR DCD and otherwise account for changes to the ABWR ECCS EM in accordance with 10 CFR § 50.46(a)(3), they are "modifications," as this term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations applicable and in effect at the initial ABWR certification.

The following regulatory requirements provide the basis for the acceptance criteria for the staff's review:

- 10 CFR § 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," provided various requirements related to the design and analysis of ECCS for light water power reactors (1997). In particular,
- 10 CFR § 50.46(a)(1)(i) required that boiling or pressurized light-water reactors be provided with an ECCS, and that the ECCS design performance be analyzed to meet certain acceptance criteria using an acceptable EM.
- 10 CFR § 50.46(a)(1)(ii) allowed ECCS EMs to be developed in conformance with the required and acceptable features of ECCS EMs described in Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (in lieu of providing a best estimate plus uncertainty EM as described in 10 CFR § 50.46(a)(1)(i)).
- 10 CFR § 50.46(b)(1) through (5) provided the ECCS acceptance criteria for peak cladding temperature (PCT), maximum cladding oxidation, maximum hydrogen generation (often expressed in terms of core wide oxidation), maintaining a coolable geometry, and providing for long term core cooling following a LOCA.
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC), provided minimum requirements for the principal design criteria for a facility. Facility principal design criteria were required under both 10 CFR Part 50 and 10 CFR Part 52. The following GDC is relevant to this particular ECCS review:
- GDC 35, "Emergency Core Cooling," which requires an ECCS, such that, following a LOCA, the ECCS transfers heat away from the core at a sufficient rate to (1) prevent fuel and clad damage that could interfere with continued effective core cooling and (2) limit the clad metal-water reaction to negligible amounts. This GDC also provided single failure criteria and electric power system requirements for the ECCS.

The applicant performed the original ABWR DCD LOCA analysis using SAFER/GESTR-LOCA, an EM approved by the NRC in 1984 (ADAMS Accession No. ML102230240 (proprietary)), for performing ECCS evaluations in accordance with 10 CFR § 50.46 and 10 CFR Part 50, Appendix K. The staff notes that by the time the original DC was approved in 1997, the ECCS rule had been updated to no longer require ECCS EMs be compliant with the required and acceptable features in Appendix K; however, GEH maintained the use of an Appendix K-compliant EM, as provided for in 10 CFR § 50.46(a)(1)(ii).

### *6.3.2 Summary of Technical Information*

In December 2010, GEH submitted an application to renew the ABWR DC. Subsequently, GEH submitted several letters[9] providing annual reports of ECCS EM changes and errors for the ABWR standard plant design, pursuant to the requirements of 10 CFR § 50.46. Some of these ABWR annual reports referenced information in earlier annual reports that had been submitted to the NRC, but had not been specifically associated with the ABWR design.

In the ABWR DCD, Revision 6, the NRC staff noted in a July 21, 2016, letter that GEH had not reported changes and error corrections in its ECCS EM, and therefore the renewal failed to meet the requirements of 10 CFR § 52.57(a), which specifies that the renewal application must contain all information necessary to bring up to date the information and data contained in the previous application. In the letter dated August 19, 2016, GEH committed to addressing the issue in the ABWR DCD, Revision 7.

By letters dated October 12, 2016, and October 10, 2017 (ADAMS Accession No. ML16291A490 and ADAMS Accession No. ML17283A307, respectively), GEH provided additional annual reports pursuant to 10 CFR § 50.46 of ECCS EM changes or errors that resulted in an increased PCT for the standard ABWR design. The October 12, 2016, annual report included additional supplemental information describing the errors and changes in more detail than the previous reports, as well as proposed changes to the ABWR DCD. The changes included the addition of a reference to the October 12, 2016, annual report, and a note on the limiting PCT result in DCD Tier 2, Table 6.3-4, "Summary of Results of LOCA Analysis," that reported the impact of the changes and errors on the PCT. In a letter dated March 20, 2017 (ADAMS Accession No. ML17079A353, with enclosures in ADAMS Accession No. ML17079A356 (Enclosure 1) and ADAMS Accession No. ML17079A357 (Enclosure 2)), GEH supplemented the information in the October 12, 2016, annual report with additional discussion and further changes to the ABWR DCD.

In August 2018, the NRC staff audited GEH information related to the PCT error and change estimates, first through an electronic portal and then in an on-site audit (see Audit Plan in (ADAMS Accession No. ML18199A273) and audit report in (ADAMS Accession No. ML19136A281). On October 29, 2018 (ADAMS Accession No. ML18302A023), the applicant provided a 2018 annual report of ECCS EM changes and errors for the ABWR pursuant to 10 CFR § 50.46. This report removed the effect of several changes and errors that GEH had previously reported. On December 19, 2018,

[9] The annual reports of ECCS EM changes/errors for the ABWR standard plant design were submitted on February 13, 2012 (ADAMS Accession No. ML12046A048); December 19, 2012 (ADAMS Accession No. ML12355A207); December 13, 2013 (ADAMS Accession No. ML13350A583); December 19, 2014 (ADAMS Accession No. ML14363A096); and December 3, 2015 (ADAMS Accession No. ML15337A119).

the NRC staff held a public meeting with GEH (ADAMS Accession No. ML19009A413) to discuss how the increase in PCT associated with the reported errors would be incorporated into the ABWR DCD. Subsequently, GEH submitted a letter on January 21, 2019 (ADAMS Accession No. ML19021A015), that justified the removal of several entries included in its October 12, 2016, annual report and provided proposed revised ABWR DCD markups.

In the January 21, 2019, letter, GEH removed several ECCS EM changes and errors from the table of cumulative PCT changes and errors that had originally been included in the October 12, 2016, annual report. After further review, the applicant found that a variety of changes and errors, including both input and modeling errors, were not applicable to the ABWR design. The sum of the remaining errors deemed applicable to the ABWR had a combined impact on the PCT of 42 degrees Celsius (°C) (approximately 75 degrees Fahrenheit (°F)).

In the ABWR DCD, markups provided as Enclosure 2 to the letter dated January 21, 2019, GEH added a new column to DCD Tier 2, Table 6.3-4, which summarizes the LOCA analysis results. This column represents a new set of licensing basis PCTs for the ABWR DC renewal and is based on adding the 42°C (75°F) value to the prior licensing basis PCT values in the table. The limiting PCT, following incorporation of estimated effects of the ECCS EM changes and errors since the original ABWR DC, is now 663°C (1225°F). The same adder of 42°C (75°F) was also applied to the LOCA evaluation results included in the internal event analysis section of the probabilistic risk assessment reported in DCD Tier 2, Section 19.3.1.3.1, “Success Criteria,” since the analyses were carried out using the same EM.

**Reported ECCS EM Changes and Errors**

As discussed above, the applicant’s original licensing basis LOCA analysis for the ABWR standard plant design was performed using the SAFER/GESTR-LOCA EM and the limiting LOCA was found to be a steam line break outside of containment. These basic concepts remain unchanged in the updated ABWR DCD, Revision 7.

Enclosure 1 to the applicant’s October 12, 2016, annual report of ECCS EM changes and errors included a table of the changes and errors in the SAFER/GESTR EM that the applicant identified since the original ABWR licensing basis analysis, including those that were found to have negligible impact. These errors were identified through continued use of the EM for operating boiling-water reactor (BWR) analyses. For each error, the applicant assessed the impact on the PCT for the ABWR. In total, the absolute value of the sum of the errors reported in the October 12, 2016, annual report was +220°F. The estimated effect on the PCT for each error depended on the nature of the individual error and how it related to the ABWR design. Of the reported errors, the applicant determined that only the error in one of the 50.46 annual reporting (AR) letters (AR Letter 2006-01) reached the 50°F criterion used in 10 CFR § 50.46(a)(3)(i) to establish that an error is significant with respect to the reporting requirements. However, the staff evaluated the errors and determined that several of the reported errors had an effect on the PCT that the NRC staff considers to be non-trivial (more than a few degrees). Others, mostly software platform ports and other minor changes, had an estimated effect of 0°F.

Enclosure 1 to the applicant’s letter dated January 21, 2019, GEH determined it was appropriate to remove several entries from the table, namely the entries included from

AR Letters 1996-01, 2002-02, 2002-04, 2006-01, 2012-01, and 2014-03. AR Letters 1996-01 and 2002-02 reported input errors related to fuel bundle modeling and inclusion of the steam dryer pressure drop in the initial core water level input. The applicant eliminated these errors in the January 2019, letter because further review found that the input was correct for the ABWR LOCA analysis. In AR Letters 2002-04 and 2014-03 related to software platform changes were removed since the analysis had not actually been performed for the ABWR, and thus, any associated changes in the PCT were not applicable to the ABWR DCD analysis.

AR Letter 2006-01 addressed the issue that it is potentially non-conservative to apply a cosine axial power shape in small break LOCA analyses and that a more top-peaked shape would have a greater impact on the PCT. This was originally assessed as applicable to the ABWR analysis because one of the ABWR LOCA sensitivity studies included with the licensing analysis was found to include what appeared to be a brief core uncovery within the first few seconds. However, on further inspection, the applicant found that the apparent uncovery was actually a departure from nucleate boiling (DNB) event caused by the depressurization and reactor internal pump trip after the initiating LOCA. During this portion of the transient, the PCT of 1,149°F is reached. Liquid droplet entrainment from the depressurization subsequently cools the fuel back to saturation conditions. GEH assessed that these phenomena differed from the event that caused the reported non-conservatism in the operating BWR fleet, and that the DNB and subsequent cooldown would occur during the ABWR LOCA transient regardless of the axial peak power location. Therefore, the applicant determined after more detailed review that the error was not applicable to the ABWR.

AR Letter 2012-01 addressed fuel thermal conductivity degradation (TCD), the process by which fuel thermal conductivity degrades with increasing irradiation. As discussed in NRC Information Notice 2009-23 (ADAMS Accession No. ML091550527), "Nuclear Fuel Thermal Conductivity Degradation," fuel performance codes the NRC approved before 1999 did not include a reduction in fuel thermal conductivity with increasing irradiation because earlier test data were inconclusive as to the significance of the effect. This is the case for the ABWR licensing analysis, which the applicant performed using the SAFER/GESTR-LOCA methodology, approved in 1984. The applicant thus initially determined that the ABWR licensing analysis did not account for TCD. Resolution of the TCD issue in the operating BWR fleet was achieved by switching from the GESTR-LOCA fuel thermal-mechanical code to a newer fuel thermal-mechanical code, PRIME. The overall effect on the PCT of implementing PRIME was initially assessed for the ABWR as 45°F. On further review, the applicant determined that TCD would not impact the LOCA analysis for the ABWR because the bounding fuel state with respect to LOCA analysis is early in core life, as discussed more fully below. The applicant thus determined that the implementation of PRIME and the associated 45°F effect on PCT was unnecessary for the ABWR.

After removing the table entries discussed above, the applicant reported the revised cumulative sum of the PCT effects from the remaining changes and errors in the January 21, 2019, letter to be 75°F.

**Effect of ECCS EM Changes and Errors on Non-LOCA Analyses**

Enclosure 1 to the applicant's March 20, 2017, letter listed analyses that had the potential to be affected by the reported ECCS EM changes and determined whether or not each analysis was in fact affected. The applicant updated the list and associated disposition in the letter dated January 21, 2019.

The applicant stated that the errors discovered in, and changes made to, the ECCS EM had no potential impact on the station blackout analysis, decay heat analysis, containment analysis, or radiological analysis. Since the changes and errors directly affected the ECCS EM, the probabilistic risk assessment (PRA) success criteria for events reported in ABWR DCD Chapter 19 that relied on the ECCS EM were also affected. Several of the originally-reported ECCS EM changes and errors had potential effects on the non-LOCA analyses, including AR Letter 1996-01, which was the error involving the incorrect number of active fuel rods in the SAFER input file, AR Letter 2003-03, which was the error involving the steam separator pressure drop, and AR Letter 2012-01, which was the change that implemented the PRIME fuel thermal-mechanical performance code to resolve the TCD issue.

The applicant stated in the January 21, 2019, letter that the AR Letter 1996-01 error was only a potential issue for the SAFER input file, which was only used in the LOCA analysis. Because the error was not present in the SAFER input file for the ABWR, the applicant determined that the error was not applicable to any ABWR analysis.

The error described in AR Letter 2003-03 could theoretically have had an impact on the transient and anticipated transient without scram (ATWS) analyses. However, the applicant determined that the transient and ATWS analysis base decks for the ABWR DCD were developed separately from the LOCA analysis base deck and confirmed that they used the correct steam separator pressure drop. Thus, the error affects only the LOCA analysis.

As discussed above, the licensee determined that the AR Letter 2012-01 change, which included the effects of fuel TCD, was not applicable to the ABWR LOCA analysis. However, the effects of TCD were not included in the transient or ATWS analyses, and GEH therefore addressed the potential impacts of TCD on these analyses. In its evaluation, the applicant referenced an NRC staff letter to GEH dated March 23, 2012," Nuclear Fuel Thermal Conductivity Degradation Evaluation for Light Water Reactors Using Ge-Hitachi Nuclear Energy Codes and Methods" (ADAMS Accession No. ML120680571), which evaluated various GEH codes and methods for operating reactors to ascertain the consequences of TCD. Enclosure 1 and 2 of that staff letter dated March 23, 2012, "Staff Assessment of General Electric Codes and Methods with Regard to Thermal Conductivity Degradation (ADAMS Accession No. ML120750001 (public version) and ML120680592 (proprietary version)), provides the staff evaluation of the codes and methods assessed included those operating plant methods that are also used for the ABWR DCD transient and ATWS analyses. The applicant, therefore, determined that the conclusions could be extended to the ABWR DCD analyses. The applicant also determined that the sensitivity to TCD for the ABWR DCD would be lower than that for the operating BWR/4 because the ABWR DCD analysis assumes an initial core with low exposure, making the effects of TCD less pronounced. The applicant therefore concluded that it is not necessary to include the effects of fuel TCD in the transient and ATWS analyses for the ABWR.

**ABWR DCD Changes**

Changes to the ABWR DCD related to the ECCS EM changes and errors were originally made in Enclosure 2 to the applicant's March 20, 2017, letter. A revised set of changes to the DCD were included in the applicant's January 21, 2019, letter, consistent with the changes to the applicant's evaluation included in the same letter.

The applicant added the 2018 annual report of ECCS EM changes and errors pursuant to 10 CFR § 50.46 as a reference in DCD Tier 2, Section 6.3.7 "Reference." The applicant also added a new column titled "Renewal PCT/w ΔPCT adjustment (°C)", and two new footnotes to DCD Tier 2, Table 6.3-4. Footnote 2 explained that the new column was added to account for errors in and changes to the ECCS EM since the NRC originally approved the ABWR DCD, and that the values in the column were determined by adding the cumulative change in PCT from the 10 CFR § 50.46 reports of 42°C (approximately 75°F) to the original ABWR DCD values for each LOCA case considered in the table. Footnote 3, which is the same as the general note for the table that was included in the approved version of the ABWR DCD, clarified the method used to calculate the core-wide metal-water reaction for the analysis.

Additionally, the applicant revised the PCTs reported in DCD Tier 2, Section 19.3.1.3.1, regarding the success criteria for the PRA of internal events. The PCTs in the section were all updated, consistent with those in DCD Tier 2, Section 6.3, by increasing each reported PCT by 42°C (75°F).

### *6.3.3 Technical Evaluation*

The NRC staff focused on three aspects of the applicant's evaluation: (1) the assessment of the reported ECCS EM errors and changes for the LOCA analysis, (2) the disposition of the same EM changes and errors for the non-LOCA analyses, and (3) the implementation of the EM changes and errors in the ABWR DCD.

**Effect of ECCS EM Changes and Errors on LOCA Analysis**

During the regulatory audit, the NRC staff reviewed the list of all errors reported in the SAFER/GESTR-LOCA EM since the ABWR LOCA analysis was originally performed in 1994. The NRC staff confirmed that the applicant's October 12, 2016, annual reporting of changes and errors pursuant to 10 CFR § 50.46 contained all items potentially applicable to the ABWR. The staff also confirmed that the applicant assigned a conservative value to the estimated effect on PCT for each change or error, based on assessments from the operating BWR fleet. The NRC staff subsequently reviewed the rationale provided in the applicant's letter dated January 21, 2019, for removing several changes and errors from consideration for ABWR, as discussed in Section 6.3.2 of this FSER supplement. Since the issues reported in AR Letters 1996-01 and 2002-02 were found to not be present in the ABWR analysis, the NRC staff concluded that it was reasonable to remove them from the list of changes and errors affecting the PCT for ABWR. Additionally, because the platform changes associated with AR Letters 2002-04 and 2014-03 were not applied in the analysis, the NRC staff determined that it was reasonable to remove those changes and errors from consideration as well.

For AR Letter 2006-01, which assessed the potential impact of using a top-peaked axial power shape in small break LOCA analyses in the operating fleet, the applicant initially concluded that the effect on PCT would be approximately 50°F, but then it revised the estimate to conclude that the effect would be negligible in the January 21, 2019, letter.

The applicant's rationale for neglecting the effect of this error is that there is no impact on the PCT because there is no core uncovery in the LOCA analysis. The applicant stated in the ABWR DCD that there are no design basis events that result in core uncovery, and the one case that the applicant observed to have potential uncovery was a sensitivity study, not a licensing basis analysis. The applicant further found that this single case represented a localized DNB event. Under these conditions, the NRC staff assessed that, while a change to the axial peak power location would potentially change the location of the DNB event, there is no phenomenological reason to believe that the duration of the heat up, and thus the PCT reached, would be extended. Therefore, the NRC staff agreed with the applicant's assessment that the error addressed in AR Letter 2006-01 would have a negligible effect on the PCT.

AR Letter 2012-01 assessed the effect on the PCT of resolving fuel TCD by implementing the PRIME fuel thermal-mechanical code. As discussed above, the applicant initially found that implementation of PRIME would change the PCT by 45°F. However, the applicant concluded in the January 21, 2019, letter that the implementation of PRIME was not necessary to solve the issue of TCD for the ABWR LOCA analysis, and therefore concluded that the change was not applicable to the ABWR standard design. The applicant's rationale is that the bounding fuel state with respect to LOCA analysis is early in core life, because of gap conductance and stored energy, and that since TCD is an irradiation effect that does not have a significant impact on fuel until later in life, it has no appreciable impact on the LOCA PCT. The NRC staff evaluated and disagrees in part with this conclusion; because experience with the issue has shown that TCD may make fuel bundles in their second cycle of operation more limiting than those in their first, depending on the arrangement of the core and other plant and cycle-specific parameters. However, since the core remains covered in the ABWR design for all break locations, the NRC staff judged that the stored energy increase resulting from TCD will be removed by increased boiling and will therefore not appreciably impact the PCT. This is consistent with the conclusion regarding small break LOCAs from the NRC staff's March 23, 2012, letter to GEH evaluating the impact of TCD for existing approved methods on operating BWR reactors. Additionally, the fuel rod design criteria specified in DCD Appendix 4B, "Fuel License Acceptance Criteria," must be satisfied for initial core and reload applications.

Therefore, based on the assessments detailed above, the NRC staff determined that the applicant adequately estimated the PCT effect of the changes and errors discovered in the SAFER/GESTR-LOCA EM since the original analysis was approved by the NRC. The total estimated effect of 42°C (75°F) is conservative and appropriate to include in the DCD to account for these changes and errors.

The NRC staff notes that, in addition to PCT, 10 CFR § 50.46(b) also contains criteria for maximum local cladding oxidation, maximum hydrogen generation, maintaining a coolable geometry, and providing for long-term core cooling following a LOCA. The NRC staff determined that none of the reported ECCS EM changes or errors would affect the ability of the ABWR to maintain a coolable geometry or to provide long term core cooling. While some of the reported changes and errors could potentially impact maximum local cladding oxidation or maximum hydrogen generation, the NRC staff judged that, because the core remains covered in the ABWR LOCA analysis, any effect would be essentially negligible and below the number of significant figures reported in the ABWR DCD.

**Effect of ECCS EM Changes and Errors on non-LOCA Analyses**

Enclosure 1 to the applicant's letter dated January 21, 2019, identifies the non-LOCA analyses in the ABWR DCD that have the potential to be affected by the ECCS EM changes and errors and provides an evaluation as to the effect on each analysis. The non-LOCA analyses identified by the applicant as potentially affected by the ECCS EM changes and errors included the station blackout analysis, the RPV fluence analysis, the decay heat analysis, the containment analysis, the combustible gas analysis, the radiological analysis, the transient analysis, the ATWS analysis, and the analysis supporting the PRA success criteria. The NRC staff evaluated and concurs with the applicant's assessment of the scope of analyses potentially affected by the ECCS EM changes and errors, since these represent the set of DCD analyses that are the most sensitive to core and fuel parameters.

The applicant reviewed the analyses listed above and found that the models used in the LOCA analysis were also used in the PRA success criteria evaluation and the break flow and mass release calculations associated with the radiological analysis. The NRC staff evaluated this assessment and confirmed this during the audit. Additionally, the applicant noted that the base input decks for the transient and ATWS analyses were developed separately from the LOCA analysis. Thus, input file errors and modeling errors specific to the LOCA EM, which represent the majority of the reported errors, would not be expected to apply to the other analyses.

Of the reported errors, only the error related to the omission of fuel thermal conductivity degradation is potentially applicable beyond the direct application of the ECCS EM. The NRC staff reviewed the applicant's justification for removing this error, which relied heavily on the NRC Staff's March 23, 2012, evaluation of the methods in use at the time. This evaluation concluded that the effect of TCD on transient and ATWS analyses would be minimal, though the GESTRM fuel rod thermal performance code could have some non-conservatism in the calculation of fuel temperatures for higher burnup fuel. The staff reviewed the transient analysis included in Chapter 15 of the ABWR DCD and found that there is sufficient margin to the fuel centerline melt limit that no events would be substantially impacted by TCD. For the rod withdrawal error during startup, analyzed in DCD Tier 2, Section 15.4.1.2, the fuel enthalpy calculations would be potentially affected by TCD, but the analysis would be expected to retain very substantial margin to the values at which fuel rod damage is anticipated. Thus, the NRC staff concludes that it is reasonable to neglect this error in the ABWR DCD non-LOCA analysis.

Changes to or errors in the ECCS EM that affect the break flow and mass and energy release would potentially affect the radiological analysis. The applicant stated in the January 21, 2019, letter that no such errors were identified. The NRC staff disagrees since AR Letters 1999-02, 2001-02, 2001-04, and 2003-03 all identified errors that have a potential impact on the mass and energy release. AR Letters 1999-02, 2001-04, and 2003-03, which identified errors in the application of counter-current flooding limitation at the top of the core, in the steam flow from the core exit, and in the steam separator pressure drop, would all be expected to impact the pressure in the vessel during the transient and thus the transient mass and energy release. AR Letter 2001-02 identified a convergence error related to the time step size; without additional details, the effect on the transient break flow and mass and energy release is difficult to estimate, but the NRC staff expects that it is non-negligible. However, the NRC staff has determined that the integrated, rather than instantaneous, break flow and mass and energy release is what is important to the radiological analysis, and these errors would not be expected to significantly influence the total amount of mass and energy ejected through the break. Thus, the NRC staff concluded that any change in the radiological analysis from the errors reported for the ECCS EM would be negligible.

For the PRA success criteria evaluation, which used the ECCS EM directly, the applicant decided to take the same approach as the LOCA analysis and apply the estimated effect of the changes and errors to the values reported in the ABWR DCD. The NRC staff finds this approach to be acceptable for addressing the issue, since the estimated effects are directly applicable to the analysis using the ECCS EM that are included in the PRA.

**ABWR DCD Changes**

The NRC staff reviewed the ABWR DCD, Revision 7, and found that they appropriately account for the changes and errors reported in the applicant's 2018 10 CFR § 50.46 annual report and the justifications provided in the January 21, 2019, letter. In the revised ABWR DCD, the applicant took the approach of accounting for the ECCS EM changes and errors discovered since the original ABWR DCD approval by adding the estimated effect of the changes and errors to the original ABWR DCD cladding temperature values. As discussed above, the NRC staff reviewed the changes and errors to the ECCS EM and found that the applicant appropriately evaluated each change or error and conservatively assessed the associated effect on the PCT.

The sum of the absolute magnitudes of the reported changes and errors is greater than 50°F, and thus exceeds the threshold of significance as defined in 10 CFR § 50.46(a)(3)(i). Nonetheless, the NRC staff evaluated this information and has determined that the reported changes and errors are relatively minor and do not call the continued acceptability of the EM into question. Additionally, none of the changes or errors impacts the features required by 10 CFR Part 50, Appendix K. By conservatively estimating the effects of the changes and errors and incorporating the estimated effects into the ABWR design basis by updating the PCT reported in the ABWR DCD, the applicant demonstrated continued compliance with the PCT acceptance criterion in 10 CFR § 50.46(b)(1).

As discussed above, the effect of the changes and errors on the maximum local oxidation also reported in the ABWR DCD is negligible.

Because the changes are relatively minor, because the applicant incorporated the estimated effects into the design basis by updating the ABWR DCD, and because the ECCS EM, including the effects of the changes and errors identified since the original ABWR DCD, demonstrates significant margin to the 2,200°F and 17 percent maximum local oxidation acceptance criteria of 10 CFR § 50.46(b)(1) and (b)(2), the NRC staff concluded that the design as modified continues to meet the requirements of 10 CFR § 50.46 and 10 CFR Part 50, Appendix K, and that the changes are therefore acceptable.

The PRA analysis in DCD Chapter 19.3 was also updated to incorporate the same change in peak cladding temperature as the ECCS evaluation in DCD Tier 2, Section 6.3. Even with the addition of 42°C (75°F) to the PCTs reported in the PRA analysis, the results maintain significant margin to the acceptance criterion established in the original ABWR DCD of 1,483°C (2,700°F). The NRC staff therefore finds the changes to be acceptable.

The staff notes that GEH chose to retain two combined license (COL) Information Items related to the ECCS performance evaluation in the ABWR DCD, Revision 7. In accordance with DCD Tier 2, Sections 6.3.6.1, “ECCS Performance Results,” and 6.3.6.3, “Limiting Break Results,” a COL applicant referencing the ABWR DCD will provide various results for the limiting break for each bundle design.

The applicant provided the necessary information in the ABWR DCD Revision 7 which incorporated the changes in response to the staff’s letter dated July 21, 2016. Therefore, Confirmatory Item 6.3-1 from the staff’s advanced safety evaluation report with no open Item for the ABWR DC renewal is resolved and closed.

### *6.3.4 Conclusion*

Based on the above, the NRC staff finds that, with the changes incorporated into the ABWR DCD, Revision 7, the ECCS performance evaluation included in the renewed ABWR DCD meet the requirements of 10 CFR § 50.46 and 10 CFR Part 50, Appendix K. Because the changes only affected the peak cladding temperature and maximum local oxidation evaluations, compliance with the 10 CFR § 50.46 acceptance criteria is sufficient to demonstrate that the design of the ECCS meets the requirements of GDC 35 and is therefore acceptable.

# 7 INSTRUCTIONAL AND CONTROL SYSTEMS

shutdown of the reactor, including necessary instrumentation and controls 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.

*7.4.1.4.4.2 Summary of Technical Information*

Item No. 26 from the staff letter dated July 20, 2012, requested that the applicant address the design related aspects of NTTF Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events as outlined in Attachment 2 of the Commission Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735).

The staff discussed NRC actions involving a pending final rulemaking for the MBDBE rule during a public teleconference held on December 1, 2016. The staff noted during that call that, according to the latest public information regarding the pending final rule, no requirements would be applicable to applicants for a standard DC (or a renewal, as in the case of the ABWR application). The staff expected the final rule to be effective before the completion of the ABWR DC renewal. On that basis, in a letter dated December 6, 2016 (ADAMS Accession No. ML16341A812), GEH informed the NRC of its plans to submit a revised response for addressing Item No. 26 by the end of January 2017. By letter dated January 23, 2017 (ADAMS Accession No. ML17025A386), the applicant provided the updated GEH response for Item No. 26, maintaining some enhanced design features related to mitigating strategies that may be used by a potential COL applicant to satisfy the MBDBE rule requirements including enhancements to the ABWR remote shutdown panel.

*7.4.1.4.4.3 Technical Evaluation*

In the letter dated January 23, 2017, the applicant stated the ABWR design enhancements provide additional features, rather than addressing specific regulatory requirements, that, for example, provide redundancy or offer operational conveniences that have been proposed by the industry. The features may be used as part of an overall approach for mitigating strategies when COL applicants or licensees implement the final MBDBE rule for the development of procedures, or programs. For these reasons, GEH elected to retain most of these design features in the ABWR DCD but did not characterize them as "mitigating strategies."

The design enhancements the applicant provided for the ABWR remote shutdown system include:

- Replacement of control for safety relief valves (SRVs) "G", "J", "K" and "P" with control for automatic depressurization system (ADS) SRVs "C", "H", "L" and "R", which can be operated by the replenishable supply of nitrogen gas (N2). This change affects DCD Tier 1, Figure 2.1.2a, and DCD Tier 2, Figure 7.3-2, Sheets 2, 3, 4, 6, 7, 9, 10 and 18, Figure 7.4-2, and Figure 7.4.3, Sheets 2 and 9.
- Addition of wide-range reactor pressure vessel water level indication (Divisions I and II) (cold calibration) to provide capability to monitor this parameter from a centralized location during extended loss of alternating current (ac) power events.

This change affects DCD Tier 1, Figure 2.1.2e, and DCD Tier 2, Sections 7.4.1.4.4, 16.3.3.6.2, and 16.B.3.3.6.2, Figure 5.1-3, sheets 5 and 6, and Figure 7.4-2.

- Addition of N2 supply header pressure indication (Divisions I and II) to provide capability to monitor this parameter from a centralized location during extended loss of ac power events. This change affects DCD Tier 1, Sections 2.2.6, 2.11.13, and Figure 2.2.6, and DCD Tier 2, Sections 7.4.1.4.4, 16.3.3.6.2, 16.B.3.3.6.2, Figure 6.7-1, and Figure 7.4-2.
- Addition of condensate storage tank water level indication (Division I, which will be in addition to the existing Division II) to provide capability to monitor this parameter from a centralized location during extended loss of ac power events. This change affects DCD Tier 1, Figure 2.11.2, and Figure 2.2.6, and DCD Tier 2, Sections 7.4.1.4.4, 16.3.3.6.2, 16.B.3.3.6.2, Figure 6.7-1, Figure 7.4-2, and Figure 9.2-4.
- Addition of containment wide-range pressure indication (Divisions I and II) to provide capability to monitor this parameter from a centralized location during extended loss of ac power events. This change affects DCD Tier 1, Figure 2.2.6, and DCD Tier 2, Sections 7.4.1.4.4, 16.3.3.6.2, 16.B.3.3.6.2, Figure 6.2-39, Sheet 3, and Figure 7.4-2.
- Addition of wide-range suppression pool water level indication (Divisions I and II) to provide capability to monitor this parameter from a centralized location during extended loss of ac power events. This change affects DCD Tier 2, Sections 7.4.1.4.4, 16.3.3.6.2, 16.B.3.3.6.2, Figure 6.2-39, Sheet 2, and Figure 7.4-2.

The applicant stated that these shutdown panel design changes will provide enhancements and additional capability for plant operation during control room evacuation as well as beyond-design-basis event conditions. The capability to operate SRVs assigned to ADS valves, that include a replenishable supply of N2 for motive force would enable operation of the ADS and SRVs from the remote shutdown panels during extended loss of ac power events such as a beyond-design-basis station blackout event. The staff finds that these design changes are enhancements to the ABWR as stated by the applicant and do not affect the staff's evaluation findings documented in NUREG–1503, Section 7.4.3 of the staff FSER for the initial ABWR DC. Specifically, the staff finding remains valid for the stated DCD amendments:

> Equipment at appropriate locations outside the control room have been provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls 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

Therefore, the staff concludes that the systems controlled from the ABWR remote shutdown panel, required for safe shutdown, satisfy the requirements of GDC 19 for capability of a prompt hot shutdown and potential capability for subsequent cold shutdown.

The applicant provided the necessary information in the ABWR DCD, Revision 7 which incorporated the changes described in the applicant's January 23, 2017 letter, Enclosure 2. Therefore, Confirmatory Item 7.4.1.4.4-1 from the staff's advanced safety evaluation report with no Open Items for the ABWR DC renewal is resolved and closed.

#### 7.4.1.4.4.4 Conclusion

The staff reviewed the GEH design enhancements as updated in the ABWR DCD, Revision 7, and determined them to be acceptable, because the changes allow for enhanced plant shutdown capabilities from the remote shutdown panels in a beyond-design-basis event such as during an extended loss of ac power. These enhanced remote shutdown system design features do not affect the staff's evaluation findings documented in NUREG–1503, Section 7.4.3 the FSER for the original ABWR DC. Therefore, the staff concludes that the systems required for safe shutdown satisfy the requirements of GDC 19 and are therefore, acceptable.

#### 7.5.2.1 Post Accident Monitoring System

In a letter dated August 25, 2015 (ADAMS Accession No. ML15237A192), the applicant proposed to add spent fuel pool (SFP) level instruments that conform with the applicable guidance specified in the Japan Lesson-Learned Project Directorate-Interim Staff Guidance (JLD-ISG)-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," dated August 29, 2012 (ADAMS Accession No. ML12221A339), which endorses with exceptions and clarifications the methodologies described in the Nuclear Energy Institute (NEI) industry guidance document NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," issued August 2012 (ADAMS Accession No. ML122400399).

Subsequently, during the pending draft mitigation of beyond-design-basis events (MBDBE) rule (10 CFR § 50.155, "Mitigation of beyond-design-basis events"), the Commission decided not to impose mitigation strategies requirements on DCs.[11] The final rule was published in the *Federal Register* on August 9, 2019 (84 FR 39684) and became effective September 9, 2019.

This change to the design of SFP instruments resulted in revisions to the ABWR DCD, Revision 7, specifically, DCD Tier 2, Section 7.5.2.1, "Post Accident Monitoring System," which incorporated safety-related SFP instrumentation to permit operators to monitor the SFP water level after an accident and to take corrective action, as necessary.
The change will also result in combined license (COL) applicants being responsible for implementing the procedures and personnel training for the SFP safety-related instrumentation. These elements are specified as part of the applicants COL Information Item 7.5.3.1, "Spent Fuel Pool Level Instruments," in the ABWR DCD, Revision 7.
In addition, this change resulted in revisions to the following ABWR DCD sections:

- Tier 1, Subsection 2.6.2, "Fuel Pool Cooling and Cleanup System," including Figure 2.6.2 and Table 2.6.2; and
- Tier 2, Chapter 1, Tables 1.8-21 and 1.8-22.

[11] In the MBDBE proposed rule regulatory analysis (ADAMS Accession No. ML15266A133), the Commission proposed to not make the MBDBE proposed rule applicable to existing DCs, which included the ABWR, because "[t]he issues that may be resolved in a DC and accorded issue finality may not include operational matters, such as the elements of the [MBDBE] proposed rule."

These ABWR design enhancements could provide a potential COL applicant the means for meeting requirements of 10 CFR § 50.155, regarding safety-related SFP instrumentation.

Section 22.2 of this FSER supplement provides the staff review of these changes and other changes associated with the new SFP instrumentation.

*7.7.1.2.1 Control Rod Ganged Withdrawal Sequence Restrictions*

*7.7.1.2.1.1 Regulatory Criteria*

In the GEH, ABWR DCD, Revision 7, the applicant corrected an error in the originally certified DCD that stated the incorrect sequence for ganged control rod withdrawal for the reactor startup evolution. This supplemental FSER evaluation documents the staff's review of the correction to the control rod ganged withdrawal sequence restrictions in DCD Tier 2, Section 7.7.1.2.1(5)(b)(iii).

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 22 of that letter, the staff asked GEH to address an apparent error in the wording of the ABWR DCD related to the ganged control rod withdrawal sequence.

Because the applicant's change is a correction to an error in the DC, it is a "modification," as this term is defined in Chapter 1 of this supplement and must comply with regulations applicable and in effect at the time the certification was originally issued. The following regulatory requirements provide the acceptance criteria for the staff's review:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 28, "Reactivity Limits," states "The reactivity control systems shall 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 reactor coolant pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold-water addition."

*7.7.1.2.1.2 Summary of Technical Information*

The originally certified DCD Tier 2, Section 7.7.1.2.1(5)(b)(iii), Revision 4, states, "Groups 1-4 may only be withdrawn before groups 5 –10 are in the full-in position." The NRC staff discovered that this statement was in error during the review of the combined license (COL) application for South Texas Project (STP) Units 3 and 4. The GEH ABWR DCD Revision 5 submitted for design certification renewal contained the same erroneous statement. The staff issued a RAI dated April 20, 2015 (ADAMS Accession No. ML15110A122), and in question 07-01 noted that if the ganged withdrawal sequence is performed as described in the DCD Tier 2, Section 7.7.1.2.1(5)(b)(iii) as cited above, the ganged control rod sequence steps could create a potentially unsafe operating condition through inappropriate limits on the amount and rate of reactivity increase. The staff concluded that the ganged withdrawal sequence, as described does not appear to comply with GDC 28 and is contrary to generally accepted BWR operating practices. The staff asked the applicant to correct the ganged withdrawal sequence description or provide a technical basis and further explanation as to why this section, as currently written, is correct and accurate. In response to the staff RAI, the applicant in a letter dated May 19, 2015 (ADAMS Accession No. ML15139A210), provided its response in Enclosure 1 and submitted DCD markups to ABWR DCD Revision 5, in Enclosure 2. GEH had subsequently incorporated this change in ABWR DCD, Revision 7. The revised language states, “Groups 1-4 must be fully withdrawn before groups 5-10 can be withdrawn from the full-in position.”

*7.7.1.2.1.3 Technical Evaluation*

In RAI question 07-01 the staff noted a wording error related to the control rod ganged withdrawal sequence in ABWR DCD Tier 2, Section 7.7.1.2.1(5)(b)(iii). In response to the staff RAI, GEH revised the wording as follows:

> Groups 1-4 must be fully withdrawn before groups 5-10 can be withdrawn from the full-in position.

The staff finds that with this correction to the wording in the ABWR DCD, the ganged control rod withdrawals will be performed in the correct sequence and the design of the rod control and information system is in compliance with GDC 28. The staff also finds that this correction to the wording in the DCD does not otherwise affect the staff’s original ABWR DC FSER documented in Section 7.7.2, “Specific Findings and Evaluations,” of NUREG–1503.

The staff verified that the ABWR DCD, Revision 7, incorporates the correction as described above. Therefore, this issue is resolved.

*7.7.1.2.1.4 Conclusion*

The applicant has corrected the ganged withdrawal sequence wording in the DCD Tier 2, Section 7.7.1.2.1(5)(b)(iii) as stated above, which the staff evaluated and finds acceptable. With this wording correction the staff finds that design of the ABWR rod control and information system is in compliance with GDC 28.

# 8 ELECTRICAL POWER

## 8.2.5 NRC Bulletin 2012-01: Design Vulnerability in Electric Power System

### *8.2.5.1 Regulatory Criteria*

This discussion pertains to the staff's evaluation of the design information in the GEH ABWR DCD, Revision 7, that addresses the vulnerability identified in NRC Bulletin (BL) 2012-01, "Design Vulnerability in Electric Power System" (ADAMS Accession No. ML12074A115). The staff issued NRC BL 2012-01, to confirm that all holders of operating licenses and combined licenses (COLs) for nuclear power reactors comply with 10 CFR § 50.55a(h)(3), and 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 17, "Electric Power Systems," or applicable principal design criteria specified in the updated final safety analysis report (FSAR). Specifically, the NRC requested licensees to provide information regarding: (1) the protection scheme to detect and automatically respond to a single-phase open circuit condition or high impedance ground fault condition on GDC 17 power circuits, and (2) the operating configuration of engineered safety feature (ESF) buses at power.

The applicant provided the ABWR DCD modifications related to the design vulnerability in the electric power system initiated by an open phase condition (OPC) to ensure compliance with the NRC regulations applicable and in effect at initial certification. Therefore, the changes are "modifications," as this term is defined in Chapter 1 of this supplemental FSER and will be evaluated by the staff using the regulations applicable and in effect at initial certification.

The following regulatory requirements provide the regulatory basis for the staff's review of the ABWR DCD Tier 1 and Tier 2, modifications to address NRC BL 2012-01.

- GDC 17 (1997), as it relates to the electric power system's (1) capacity and capability to permit functioning of structures, systems, and components (SSCs) important-to-safety, (2) independence, redundancy, and availability, (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, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies
- 10 CFR § 52.47(a)(1)(vi) (1997), "Contents of applications," states that an application for design certification must contain proposed inspections, tests, analyses, and acceptance criteria (ITAAC) necessary and sufficient to provide reasonable assurance that, if the tests, inspections and analyses are performed and the acceptance criteria met, a plant that references the design is built and will operate in accordance with the design certification

The purpose of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Branch Technical Position (BTP) 8-9, "Open Phase Conditions in Electric Power System," issued July 2015 (ADAMS Accession No. ML15057A085), is to provide guidance to the staff in reviewing various licensing actions of electric power system design vulnerability from OPCs in offsite electric power systems in accordance with GDC 17 or principal design criteria specified in the FSAR, 10 CFR § 50.36(c)(2) and 10 CFR § 50.36(c)(3), 10 CFR § 50.55a(h)(2), and 10 CFR § 50.55a(h)(3).

The guidance in BTP 8-9, related to offsite power systems, has the following criteria:

- Automatic detection of the loss of one or two of the three phases of the independent circuits on the high voltage side of a transformer connecting an offsite power circuit to the transmission system under all operating electrical system configurations and loading conditions: with a high impedance ground fault condition; and without a high impedance ground fault condition; and
- The automatic alarm in the main control room (MCR) under all operating electrical system configurations and plant loading conditions.

*8.2.5.2 Summary of Technical Information*

The applicant added design features and ITAAC associated with the detection, alarm and response to OPC and unbalanced phase condition (UPC) in the offsite power systems. Specifically, the modifications include: (1) DCD Tier 1, Section 2.12.1, "Electric Power Distribution System," description for monitoring, detection, alarm and response to an OPC in the offsite power system, (2) DCD Tier 1, Table 2.12.1, ITAAC Item Nos. 28, and 29; for verification that OPC and UPC are detected by non-safety-related relays for a designated relay setpoint and that a response is initiated, (3) DCD Tier 2, Table 1.9-1, COL Information Items 8.16 and 8.17 for the COL applicant to develop procedures and train operators on how to detect OPC at the main power transformer (MPT), unit auxiliary transformers (UATs), and reserve auxiliary transformer (RAT), and (4) DCD Tier 2, Section 8.1.2.2.1, "Monitoring and Protection Against Design Vulnerabilities," which explains that the ABWR standard plant design incorporates the requirements for mitigation of OPC as identified in BL-2012-01.

*8.2.5.3 Technical Evaluation*

The scope of the evaluation in this section is limited to the detection and alarms, as described in the guidance outlined in BTP 8-9, for the offsite power system. Supplemental FSER Section 8.3.3.17 discusses the mitigation aspects of OPC protection as described in BTP 8-9, for the onsite Class 1E power system.

**Offsite System OPC Detection and Alarm - MPT, UATs and RAT**

The staff reviewed the ABWR DCD, Revision 7, including responses to requests for additional information (RAIs) and DCD Tier 1 and Tier 2 modifications to the electrical system design to ensure that the design includes features to automatically detect and alarm in the MCR in response to an OPC event and is consistent with the guidance in BTP 8-9.

In a letter dated December 7, 2010 (ADAMS Accession No. ML110040176), GEH submitted ABWR DCD, Revision 5, for renewal of the ABWR DC. The ABWR DCD, Revision 5, did not include information related to Bulletin 2012-01. Following the issuance of BL 2012-01, the staff requested that GEH provide additional information to ensure that the applicant addressed the OPC issues identified in BL 2012-01, as part of the DC renewal. Therefore, in RAI 08.02-1 dated April 24, 2014 (ADAMS Accession No. ML14114A566), and in RAI 08.02-2 dated June 9, 2015 (ADAMS Accession No.

ML15154B692), the staff requested that the applicant provide the design details of OPC detection and protection schemes and how they met the requirements specified in GDC 17 and 10 CFR § 50.55a(h)(3). Specifically, the staff requested the applicant to provide design features that would (1) automatically detect OPC and alarm in the MCR under all operating electrical system configurations and (2) automatically transfer safety-related buses to alternate offsite power source or onsite standby power system within the time assumed in the accident analysis due to an OPC. In addition, the staff requested that the applicant provide associated ITAAC to ensure that OPC monitoring, detection, alarm and automatic transfer of safety-related buses to the alternate source is accomplished when an OPC occurs.

The applicant responded to RAI 08.02-1 on August 29, 2014, (ADAMS Accession No. ML14241A556). In its response, the applicant stated, in part, that detection of OPC is alarmed in the MCR so that operators can take manual action, as appropriate, and initiate corrective actions to address the loss of phase condition. BL 2012-01 includes guidance for protection systems to automatically initiate protective actions without manual actions as required by 10 CFR § 50.55a(h)(3). Since the response to RAI 08.02-1 described manual actions when an OPC is detected, the staff issued RAI 08.02-2 requesting the alarm and automatic response to an OPC and UPC. In the response to RAI 08.02-2 (ADAMS Accession No. ML15271A170), the applicant proposed design features to automatically detect OPC and UPC and alarm in the MCR, under all operating electrical system configurations and plant loading conditions. The applicant also proposed DCD Tier 1, Section 2.12.1, "Electrical Power Distribution System," ITAAC Item Nos. 26 through 30 to ensure that both OPC and UPC can be detected and alarmed in the MCR, and that the safety-related buses can be automatically separated from the offsite power source and transfer safety-related loads to the un-affected offsite power source or the emergency diesel generators when an OPC or UPC occurs.

The applicant supplemented its response to RAI 08.02-2 in letters dated May 24, 2016 (ADAMS Accession No. ML16145A346), and December 14, 2016 (ADAMS Accession No. ML16349A171) to provide additional information, clarification, and updates to the ABWR DCD. Also, in the December supplemental response to RAI 08.02-2, the applicant replaced DCD Tier 1, Table 2.12, ITAAC Item Nos. 26 through 30 with revised ITAAC Items Nos. 28, 29, and 30 to address OPC and UPC, and deleted DCD Tier 1 Table 2.12, ITAAC Items 26 and 27, which are shown as deleted items in DCD Tier 1, Table 2.12.1. The following includes the GEH changed design features and ITAAC:

ABWR DCD design features to detect and alarm in the MCR:

- Non-safety-related relays on the primary and secondary side of the MPT are designed to monitor OPC. Alarm is initiated in the MCR if OPC conditions are detected.
- Non-safety-related relays on the primary and secondary sides of the UATs and RAT are designed to automatically sense the loss of a single phase (or multiple phases) and loss of phase with ground during all plant operating scenarios and loading conditions. Alarm is initiated in the MCR if OPC conditions are detected.
- Non-safety-related relays on the primary and secondary sides of the UATs and RAT that automatically sense an unbalanced phase during all plant operating scenarios and loading conditions. Alarm is initiated in the MCR if UPCs are detected.

ABWR DCD Revised ITAAC:

- DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, to verify that the non-safety-related microprocessor based protective relays at the MPT, UATs, and RAT upon detection of OPCs, will: (1) alarm in the MCR; (2) trip or fast transfer, the non-safety-related buses.
- DCD Tier 1, Table 2.12.1, ITAAC Item No. 29 to verify the non-safety-related microprocessor based protective relays located on the feeders from offsite to the safety-related buses will: (1) detect unbalanced phase condition (UPC); (2) send an alarm to the MCR; and (3) send a trip signal to open the non-safety-related circuit breakers.
- DCD Tier 1, Table 2.12.1, ITAAC Item No. 30 to verify that the safety-related microprocessor based protective relays located on the safety-related buses will: (1) trip or fast transfer, power to alternate non-safety-related power source if the alternate power source is available; or (2) isolate the safety-related bus, shed safety-related loads, start the safety-related emergency diesel generator if no alternate source is available.

The DCD Tier 1, Table 2.12.1, ITAAC Item No. 29 and ITAAC Item No. 30 are evaluated in Section 8.3.3.17 of this FSER supplement. Additionally, in the response to RAI 08.02-2, the applicant indicated that the ABWR design follows the guidance in BTP 8-9, as stated in DCD Tier 2, Table 1.8-19, “Standard Review Plans and BTP Applicable to ABWR.”

In summary, the applicant provided design features that would automatically detect OPCs and alarm in the MCR under all operating electrical system configurations and plant loading conditions. The design features include the addition of non-safety-related relays on the primary and secondary sides of the MPT, UATs and the RAT to automatically detect and alarm in the MCR when OPC occurs. ABWR DCD Tier 2, Section 8.3.1.0.6.3, “Bus Protection,” describes the design features of the relays on the MPT, UATs, and RAT that include automatic sensing for loss of a single phase (or multiple phases) and loss of phase with ground during all plant operating scenarios and loading conditions, and alarm in the MCR. In DCD Tier 1, Section 2.12.1, and Table 2.12.1, describe the design features to detect and alarm an OPC as discussed above.

In addition, implementation of protection features for OPC and UPC offsite power systems would be adequately addressed by providing an ITAAC to verify that the detection/alarm is constructed in accordance with the design. Furthermore, the procedures and the training for the detection/alarm scheme should provide assurance that the electrical power system will address the loss of one or more of the three phases of the offsite power circuit during the life of the plant. These steps would ensure that with adequate capacity and capability, the ac power from the offsite power system would be available to safety-related equipment to meet the intended safety functions in accordance with GDC 17 requirements.

Since the MPT, UATs, and RAT have non-safety-related relays on the primary and secondary sides to automatically detect and alarm in the MCR when OPC occurs, this design feature satisfies the BTP 8-9 criterion for automatic detection and the triggering of an alarm in the MCR upon detection of an OPC. Therefore, the staff finds that the

ABWR OPC detection and alarm design as described in the ABWR DCD, Revision 7, is acceptable and conforms to BTP 8-9. The staff evaluation of the ITAAC is further discussed below.

**Offsite System UPC Detection and Alarm - UATs and RAT**

The applicant's response to RAI 08-02, GEH explained that the design features include additional capabilities to detect UPC at the UATs and RAT. It is important to note that the UPC is an additional feature provided by the applicant that is outside the scope of BTP 8-9. A UPC will be automatically detected and alarmed in the MCR under all operating electrical system configurations and plant loading conditions. DCD Tier 2, Section 8.3.1.0.6.3, explains that the relays on the primary and secondary sides of the RAT and the UATs are used to monitor UPCs in any combination on all three phases. Alarms in the MCR alert the operator to an abnormal condition. Therefore, the staff finds that the UPC design described in the ABWR DCD, Revision 7, which includes the UPC detection capabilities at the UATs and RAT and alarm in the MCR, is acceptable.

**ITAAC for Offsite System OPC Detection and Alarm**

The staff finds that the use of DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, to verify that design features associated with detection and alarming of OPC in the MCR described in the ABWR DCD, Revision 7, is acceptable. Specifically, the design commitment of ITAAC Item No. 28, verifies that the non-safety-related relays on the MPT, UATs, and RAT will be able to detect OPC or faults, then trigger an alarm in the MCR and transfer to non-safety-related buses. A COL applicant is required by ITAAC Item No. 28, to perform a test of the as-built relays on the MPT, UATs, and RAT to ensure that OPC and faults can be detected and alarmed in the MCR. The staff finds that the ITAAC will confirm that the relays used to detect OPC can detect OPC in any combination of the three phases and demonstrate the relay setpoints are set according to the setpoint methodology. The setpoint methodology has been evaluated by the staff in Section 7.2.7 of NUREG–1503, the staff FSER for the original ABWR DC. In addition, DCD Tier 2, Table 1.9-1, "Summary of ABWR Standard Plant COL License Information," COL Information Items 8.16, "Mitigation of Open Phase Condition on RAT and UATs," and 8.17, "Mitigation of Open Phase Condition on Main Power Transformer (MPT)," are provided in DCD Tier 2, Sections 8.3.4.10 and 8.3.4.11, requires the COL applicant to develop procedures and train operators on how to respond to MCR alarms and protective actions indicating abnormal conditions including OPC on the MPT, RAT, and UATs. The staff finds that DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, is acceptable because the COL applicant is required to verify that the as-built design can automatically detect an OPC at the high side of the transformers (MPT, UATs, and RAT), and alarm in the MCR, when an OPC occurs.

The applicant provided the necessary information in the ABWR DCD, Revision 7, which incorporated the changes in the applicant's responses to RAI 08.02-1 and RAI 08.02-2. Therefore, Confirmatory Item 8.2-1 from the staff's advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*8.2.5.4 Conclusion*

The staff finds that the ABWR DCD descriptions and design modifications are acceptable because they conform to the guidance in BTP 8-9 for automatic detection and alarm of OPC and therefore, meet the requirements in GDC 17 (1997), for the offsite electric power system to ensure the proper functioning of SSCs important-to-safety, and the ITAAC Item No. 28, meet the requirement in 10 CFR § 52.47(a)(1)(vi) (1997), to ensure that the design changes will be constructed and operated based on design changes reflected in the ABWR DCD, Revision 7.

### *8.3.3.17 NRC Bulletin 2012-01: Design Vulnerability in Electric Power System*

#### *8.3.3.17.1 Regulatory Criteria*

This supplemental FSER section discusses the staff's evaluation of the design information in the GEH ABWR DCD, Revision 7, that addresses the vulnerability identified in BL-2012-01 has been evaluated. As discussed in Section 8.2.5, of this FSER supplement, the staff issued BL-2012-01, to confirm that all holders of operating licenses and COLs for nuclear power reactors comply with 10 CFR § 50.55a(h)(3), and GDC 17 or principal design criteria specified in the FSAR. Specifically, the NRC requested licensees to provide information on (1) the protection scheme to detect and automatically respond to a single-phase open circuit condition or high impedance ground fault condition on GDC 17 power circuits, and (2) the operating configuration of ESF buses at power.

The applicant updated the ABWR DCD, Revision 7, with modifications related to the design vulnerability in the electric power system initiated by an OPC to ensure compliance with NRC regulations applicable and in effect at initial certification. Therefore, the updated design changes are "modifications," as that term is defined in Chapter 1 of this FSER supplement and will be evaluated using the regulations applicable and in effect at the initial ABWR certification.

The following requirements provide the regulatory basis for the staff's review of the ABWR DCD Tier 1 and Tier 2, modifications to address NRC BL 2012-01.

- GDC 17 (1997) "Electric Power Systems," as it relates to the electric power system's (1) capacity and capability to permit functioning of SSCs important to safety, (2) independence, redundancy, and availability, 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, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.
- 10 CFR § 50.55a(h) (1997), "Codes and Standards – Protection Systems," requires that for construction permits issued after January 1, 1971, protection systems must meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) "Criteria for Protection Systems for Nuclear Power Generating Stations," (IEEE Std. 279) in effect on the formal docket dates of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE Std. 279 that become effective.
- 10 CFR § 52.47(a)(1)(vi) (1997), "Contents of applications," which states that an application for design certification must contain proposed ITAAC criteria which are necessary and sufficient to provide reasonable assurance that; if the tests,

inspections and analyses are performed and the acceptance criteria met, a plant that references the design is built and will operate in accordance with the design certification.

Acceptance criteria adequate to meet the above regulatory requirements include:

- Regulatory Guide (RG) 1.75, “Physical Independence of Electric Systems,” Revision 2, September 1976, as it relates to the isolation between Class 1E buses and loads designated as non-Class 1E.
- IEEE Std. 279-1971, “IEEE Standard:  Criteria for Protection Systems for Nuclear Power Generating Stations,” as it relates to Class 1E protection systems.
- IEEE Std. 308-1980, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations,” as it relates to components, equipment, or systems used to provide isolation protection.
- IEEE Std. 384-1981, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” as it relates to the separation of Class 1E and non-Class 1E circuits.

The purpose of BTP 8-9 is to provide guidance to the staff in reviewing various licensing actions related to electric power system design vulnerability due to OPCs in offsite electric power systems in accordance with GDC 17 or principal design criteria specified in the FSAR, 50.36(c)(2), 10 CFR § 50.36(c)(3), 10 CFR § 50.55a(h)(2), and 10 CFR § 50.55a(h)(3).

The ABWR design was approved based on 10 CFR § 50.55a(h) (1997), which requires that protection systems meet the IEEE Std. 279 requirements.  BTP 8-9 states in part that protection scheme should comply with applicable requirements including 10 CFR § 50.55a(h)(2), which require compliance with IEEE Std. 279-1971 or IEEE Std. 603–1991, ‘‘Criteria for Safety Systems for Nuclear Power Generating Stations.’’ Therefore, both 10 CFR § 50.55a(h) (1997) and 10 CFR § 50.55a(h)(2) includes the same requirement for the protection systems, in that both regulatory requirements require that the protection systems meet the requirements in IEEE Std. 279-1971.

*8.3.3.17.2 Summary of Technical Information*

The ABWR DCD modifications include (1) DCD Tier 1, Section 2.12.1, “Electric Power Distribution System,” description of the safety-related design features used to protect the safety-related buses, (2) DCD Tier 1 Table 2.12.1, ITAAC Item No. 30, for verification that the safety-related relays can protect the safety buses when an OPC occurs at designated relay setpoint and (3) DCD Tier 2, Section 8.3.1.0.6.3, “Bus Protection,” and DCD Tier 2, Section 8.3.1.1.6.3, “Bus Protection,” provides description of the bus protection scheme in response to an OPC.

*8.3.3.17.3 Technical Evaluation*

The scope of the evaluation in this section is limited to the mitigation aspects of OPC protection as described in BTP 8-9, for the onsite Class 1E power system.  DCD Tier 2, Section 8.2.5 discusses the aspects regarding detection, and alarms, as described in the guidance outlined in BTP 8-9, for the offsite power system.  In addition, DCD tier 2, Section 8.2.5 provides information about the RAIs associated with BL 2012-01 for the

OPC. The review of this section associated with the protection features to provide a response to an OPC is to determine whether the design features comply with the 10 CFR § 50.55a(h) (1997) and GDC 17 (1997) requirements, conforms with BTP 8-9, and whether the applicable ITAAC meets the requirements in 10 CFR § 52.47(a)(1)(vi) (1997).

**Safety-Related Protection Features**

The staff reviewed the applicant's responses to RAI 08.02-02, supplemented in letters dated May 24, 2016, and December 14, 2016, and the ABWR DCD, Revision 7 (Tier 1 and Tier 2), modifications to the ABWR electrical system design to ensure that the design includes features to protect safety-related systems so that power can be transferred from offsite power source to the onsite power sources due to an OPC event. In the discussion provided below, the staff also reviewed the OPC modifications to ensure that electrical isolation between safety and non-safety-related systems were maintained.

The design features incorporated in the ABWR DCD Revision 7, to protect the safety-related systems from OPC, include a safety-related bus protective relay controlling the safety-related circuit breaker. In its response to RAI 08.02-2 dated December 14, 2016, the applicant explained that the safety-related relay controlling the safety-related circuit breaker will automatically separate the safety-related bus from the non-safety-related bus fed by the UAT normal preferred power with detection of OPC or ground faults. The applicant's response, included DCD markups to DCD Tier 2, Section 8.3.1.1.6.3, "Bus Protection," that stated that the bus protection scheme automatically senses loss of a single, or multiple phases, and loss of phase with ground during all plant operating scenarios and loading conditions. In addition, the safety-related relays include design features to detect UPCs. The applicant has incorporated all these changes into the ABWR DCD, Revision 7.

The guidance in BTP 8-9 states that if offsite power circuit(s) is (are) functionally degraded due to OPCs, and safe-shutdown capability is not ensured, then the ESF buses should be designed to transfer automatically to the alternate reliable offsite power source or onsite standby power system within the time assumed in the accident analysis and without actuating any protective devices, given a concurrent design basis event. In the response to RAI 08.02-2, dated December 14, 2016, the applicant stated that the safety buses are normally loaded such that a fault (including a phase loss) is detected. The staff notes that for a normally loaded bus it is easier to detect a fault (including a phase loss) than for a lightly loaded bus due to the sensitivity of the protection system relays; detecting OPC in a lightly-loaded bus would require sensitivity for lower currents. Further, in the applicant's response to RAI 08.02, GEH explained that the two safety-related buses, normally connected to UATs, will fast transfer at the safety bus level. If the fast transfer is successful, the safety electrical loads will be sequenced to the RAT. If the fast transfer is not successful, the emergency diesel generators (EDGs) will be started automatically and the safety-related electrical loads will be sequenced on to the safety-related buses as part of the EDG loading sequence. In addition, the applicant stated that the above will occur within the time frame assumed in the accident analysis and without actuating any unnecessary protective devices, given a concurrent design basis event.

The modified OPC description in DCD Tier 1, Section 2.12.1, “Electrical Power Distribution System,” describes the isolation between the safety-related and non-safety-related electric power systems and states that the electric power to safety-related buses is provided through two feeder circuit breakers (one Class 1E and one non-Class 1E) in series. RG 1.75, Revision 2, which endorses IEEE Std. 384-1981 and IEEE Std. 308-1980 for circuit breakers or fuses that are automatically opened by fault current, which specifies that Class 1E breakers are an acceptable method for isolation between the Class 1E and the non-Class 1E systems. The staff evaluated the modification and finds that the safety-related breakers, which are in series with the non-safety breakers provides separation between safety-related and non-safety systems in the two-breaker scheme. Therefore, the staff determined that the safety-related breakers provide adequate separation between the safety-related and non-safety systems and satisfies the guidance described in RG 1.75, Revision 2.

Additionally, in RAI 08.02-02, the staff requested the applicant explain how the design addresses a protection scheme to demonstrate compliance with applicable requirements including single failure criterion for safety-related systems as specified in GDC 17, and 10 CFR § 50.55a(h)(3). 10 CFR § 50.55a(h)(3) requires compliance with IEEE Std. 603-1991, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” as endorsed by RG 1.153, “Criteria for Safety Systems,” Revision 1, June 1996. The staff asked the RAI with respect to 10 CFR § 50.55a(h)(3), but 10 CFR § 50.55a(h) (1997) is applicable for the GEH ABWR design pertaining to the protection systems, which are required to meet the IEEE Std. 279-1971 requirements. In addition, RAI 08.02-2 requested that the applicant explain how the safety-related protection system design addresses a single failure due to OPC or failure in the non-safety-related protection system, such that the safety-related system is not prevented from performing its intended safety function.

In its response to RAI 08.02-2, the applicant explained that the design conforms to the IEEE Std. 603 single failure criterion because the safety-related protective relays and safety-related sequencing logic on each of the three safety-related buses are independent of those on the other safety-related buses. The staff notes that both IEEE Std. 603-1991 and IEEE Std. 279-1971 establish the single failure criterion for protection systems. The staff evaluated the response to RAI 08.02-2 pertaining to the single failure criterion, based on meeting the requirements in IEEE Std. 279-1971. IEEE Std. 279-1971, states in part that the protection system shall automatically initiate appropriate protective action, and that any single failure within the protection system shall not prevent proper protective action at the system level. The staff evaluated the information and finds that the safety-related protective relays on the safety buses satisfies the IEEE Std. 279 requirements for ensuring that any single failure within the protection system will not impact the protection system actions, since the safety-related buses and the respective protective relays are independent of each other. The staff finds that the design satisfies the IEEE Std. 279-1971 single failure criterion requirements for the OPC protection scheme, and therefore complies with 10 CFR § 50.55a(h) (1997) for safety systems.

The applicant added DCD Tier 1, Table 2.12.1, ITAAC Item No. 30, to verify the safety-related protective relays located on the safety-related buses to protect against loss of phase(s) condition. Specifically, the design commitment of ITAAC Item No. 30, verifies that safety-related relays will protect against OPC by transferring to an alternate source. The DCD Tier 1, Table 2.12.1, ITAAC No. 30, requires the COL applicant to perform a test of the as-built safety-related relays. The established relay setpoint is used to ensure that a transfer to the alternate power source or onsite source is accomplished when on OPC occurs. The staff evaluation of the ITAAC is discussed in the section titled "ITAAC for the Transfer Alternate Offsite Power Source," in this SER section supplement below.

Since the ABWR design to mitigate OPC includes features to protect safety-related systems so that power can be transferred from offsite power source to the onsite power sources due to an OPC with or without ground fault conforms to the guidance in BTP 8-9, provides adequate separation between the safety and non-safety systems satisfying the guidance in RG 1.75, Revision 2, and satisfies the single failure requirements in IEEE Std. 279-1971. Therefore, the staff finds the ABWR OPC design acceptable with respect to the OPC mitigation.

**Technical Specifications**

Regarding the testing of the safety-related protection features during the operation of the plant, the certified ABWR DCD technical specifications (TS) surveillance (SR) 3.3.8.1.3, requires the performance of a system functional test, which demonstrates that the safety-related relays can actuate at the prescribed setpoint. The setpoint methodology, as discussed in Section 8.2.5 of this supplemental FSER, will establish the setpoints for the safety-related relays used for protection against OPC. In addition, the safety-related relays will be tested to ensure that relays are able to protect the safety-related buses against an OPC. Thus, the methodology for determining the setpoints for the safety-related relays for protection against OPC is established in the ABWR DCD, Revision 7. In addition, TS SR 3.3.8.1.3 requires performance of a system functional test to demonstrate system actuation from a simulated or actual signal. Therefore, the staff finds that the safety-related protection features will be tested per the applicable TS requirements, and the setpoints are established based on the methodology described in the ABWR DC, and therefore, are acceptable.

**Non-Safety-Related Protection Features**

The non-safety-related protection design features includes non-safety-related relays which are located at the MPT, UATs, and RAT. The OPC detection features of the non-safety-related relays protects the safety buses by isolating the incoming feeders through the opening of the non-safety feeder breakers, which are in series with the safety-related feeder breakers. Therefore, power is disconnected to the safety-related buses by opening the non-safety circuit breaker(s). DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, will then be used to verify that upon a detection of OPC or fault at the transformers, a trip or a fast transfer of the non-safety-related buses to the alternate power source (RAT) will occur. As discussed in the response to RAI 08.02-2, in the event of a fault including loss of phase, the safety buses on the UATs will fast transfer to the RAT, and the fast bus transfer is alarmed in the MCR. If the fast transfer is successful, the safety electrical loads will be sequenced to the RAT. If the fast transfer is not successful, the EDGs will be started automatically and the safety electrical loads will be sequenced on to the safety-related buses as part of the EDG loading sequence as described in the ABWR DCD Chapter 8. The applicant also explained that the sequence of events discussed above will occur within the time frame assumed in the accident analysis and without actuating any unnecessary protective devices, given a concurrent design basis event. The staff evaluated and finds this aspect of the design acceptable because the implemented design features detect an OPC, provide an alarm in the control room, and ensure power is provided from either the RAT or the EDGs, and, therefore, meets the guidance in BTP 8-9.

**ITAAC for the Transfer Alternate Offsite Power Source**

DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, will be used to verify that upon a detection of OPC or fault at the transformers, a trip or a fast transfer of the non-safety-related buses to the alternate power source (i.e., RAT) will occur. In ITAAC Item No. 28, the COL applicant is required to perform a test of the as-built MPT, UAT, and RAT non-safety-related relays at designated setpoints. This design configuration and ITAAC will verify when an OPC is detected, that a trip or a fast transfer of the non-safety-related buses to the alternate power source (RAT) will occur. The staff evaluated this design configuration and finds that DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, is acceptable because the COL applicant will be required to verify that the as-built non-safety relays will detect OPC, trip or fast transfer, of the non-safety-related buses to the alternate power source, when an OPC occurs.

**ITAAC for the Mitigation of UPC**

DCD Tier 1, Table 2.12.1, ITAAC Item No. 29 will be used to verify the non-safety-related protective relays located on the feeders from offsite to the safety-related buses will: (1) detect UPC, (2) send an alarm to the MCR, and (3) send a trip signal to open the non-safety-related circuit breakers. The inspection, test and analyses of ITAAC Item No. 29, require the COL applicant to perform a test of the as-built non-safety-related relays for UPC at designated setpoints. The staff evaluated and finds that this design configuration and ITAAC will verify when UPC is detected, that the non-safety-related feeders are disconnected by the opening of the non-safety feeder breakers. The staff also finds that DCD Tier 1, Table 2.12.1, ITAAC Item No. 29, is acceptable because the COL applicant will be required to verify that the as-built non-safety relays will detect UPC, alarm in the MCR, and open the power feeder breakers, when a UPC occurs.

**TAAC for the Onsite System Mitigation of OPC**

As discussed in Section 8.2.5 of this FSER supplement, the DCD Tier 1, Table 2.12.1, ITAAC Item No. 28, will verify that upon a detection of OPC or fault at the transformers, a trip or a fast transfer of the non-safety-related buses to the alternate power source (i.e., RAT) will occur. In the applicant's response to RAI 08.02-2, GEH explains that in the event of a fault including loss of phase, the safety buses on the UATs will fast transfer to the RAT, and the fast bus transfer is alarmed in the MCR. If the fast transfer is successful, the safety electrical loads will be sequenced to the RAT. If the fast transfer is not successful, the EDGs will be started automatically and the safety electrical loads will be sequenced on to the safety-related buses as part of the EDG loading sequence as described in the ABWR DCD Chapter 8. The applicant also explained that the sequence of events discussed above will occur within the time frame assumed in the accident analysis and without actuating any unnecessary protective devices, given a concurrent design basis event. The staff evaluated this information and finds this ITAAC acceptable because it will be used to verify that the relays can detect OPC and initiate a trip, or fast transfer, to an alternate source at the designated relay setpoint.

DCD Tier 1, Table 2.12.1, ITAAC Item No. 30, is used to verify that safety-related protection relays which control the normal and alternate feeder circuit breakers are able to protect the safety-related loads against loss of phase(s) conditions. DCD Tier, Table 2.12.1, ITAAC Item No. 30, requires the performance of a test on the safety-related protective relays to demonstrate that at the designated relay setpoint, the relays will automatically: (1) trip the safety-related circuit breakers or fast transfer, if the alternate power source is available, or (2) start and transfer loads to the EDG if the alternate power source is unavailable. The staff evaluated this information and finds that DCD Tier 1, Table 2.12.1, ITAAC Item No. 30, is acceptable because the COL applicant will be required to verify that the as-built relay design automatically transfers the safety-related loads to the alternate source or EDG when an OPC occurs.

The applicant provided the necessary information in the ABWR DCD, Revision 7, which incorporated the appropriate changes described in the applicant's responses to RAI 08.02-02. Therefore, Confirmatory Item 8.3.3.17-1 from the staff advanced SER with no Open Item for the ABWR DC renewal is resolved and closed.

#### *8.3.3.17.4 Conclusion*

The staff finds that the design modifications, to ABWR DCD Revision 7, to add safety-related protection relays to protect against OPC, including ITAAC, and descriptions conform to the guidance in BTP 8-9 as it relates to the protection features to mitigate and provide a response to the OPC event, and hence, complies with GDC 17 (1997) as it pertains to OPC. The staff also finds that the ABWR DCD Revision 7 OPC design complies with 10 CFR § 50.55a(h) (1997) for safety systems, since the relays to mitigate OPC events, are separate and independent for each safety-related division.

### *8.3.4.4 Isolation Between Class 1E Buses and Loads Designated as Non-Class 1E*

#### *8.3.4.4.1 Regulatory Criteria*

In the GEH ABWR DCD Revision 7, the applicant completed a design change to add non-safety reactor building (RB) external connections for providing electrical power to the safety-related 480-volt (V) alternating current (VAC) RB Class 1E power centers from an external power source. These additional 480-V electrical connections to the safety-related 480-V system would improve the diversity and defense in depth during beyond-design-basis events and could provide a potential combined license (COL) applicant the means for meeting the requirements of 10 CFR § 50.155, "Mitigation of beyond-design basis events" (the MBDBE rule).

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. The applicant was requested by the staff in Item No. 26, of the July 20, 2012, letter to address ABWR DCD design changes related to aspects of the NRC Fukushima Near Term Task Force Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events based on the NRC policy, at that time, which was outlined in a staff requirements memorandum (ADAMS Accession No. ML120690347) for SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A111).

Subsequently, during the MBDBE rulemaking that created 10 CFR § 50.155, the Commission decided not to impose mitigation strategies requirements on DCs.[12] In a letter dated January 23, 2017 (ADAMS Accession No. ML17025A386), GEH provided supplemental information in response to Item No. 26 of the NRC's suggested ABWR design changes. The applicant narrowed the scope of Item No. 26 to exclude changes directly related to SECY-12-0025, pending final rulemaking for the MBDBE rule. GEH retained the related design change of non-safety RB external connections to provide electrical power to the safety-related 480 VAC RB 1E power centers from an external power source as an operational enhancement to provide additional defense in depth. These ABWR design enhancements could provide a potential COL applicant the means for meeting the MBDBE rule.

These changes do not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), these design changes are "amendments," as this term is defined in Chapter 1 of this supplement and will correspondingly be evaluated using the regulations in effect at renewal. In this case, the change made by GEH was not required by the regulations, but for the purposes of evaluating the applicant's DCD design amendments to add RB external connections for providing electrical power to the safety related 480 VAC RB 1E power centers, the staff evaluated the change to ensure consistency with the following regulatory requirement and associated guidance:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants,"
- (GDC) 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The independence of safety-related equipment and circuits, and auxiliary supporting features is established and maintained via physical separation and electrical isolation.
- Regulatory Guide (RG) 1.75, "Physical Independence of Electric Systems," Revision 2, issued September 1978 (ADAMS Accession No. ML003740265), provides guidance addressing independence and specifically, physical separation and electrical isolation.

*8.3.4.4.2 Summary of Technical Information*

Item No. 26 from the staff letter dated July 20, 2012, requested that the applicant address the design related aspects of Fukushima Near-Term Task Force Recommendation 4.2 mitigation strategies for beyond-design-basis external events as outlined in Attachment 2 of the Commission Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735).

As described in the version of the draft final MBDBE rule that was publicly available in November 2016, no requirements would be applicable to applicants for a standard DC

[12] In the MBDBE proposed rule regulatory analysis (ADAMS Accession No. ML15266A133), the Commission proposed to not make the MBDBE proposed rule applicable to existing DCs, which included the ABWR, because "[t]he issues that may be resolved in a DC and accorded issue finality may not include operational matters, such as the elements of the [MBDBE] proposed rule."

(or a renewal, as in the case of the ABWR application). It was also expected, at that time, that the final rule would be effective before the ABWR DC renewal would be completed.[13] On that basis, in a letter dated December 6, 2016 (ADAMS Accession No. ML16341A812), GEH informed the NRC of its plans to submit a revised response for addressing Item No. 26 by the end of January 2017. In its January 23, 2017, letter the applicant provided the updated GEH response for Item No. 26, maintaining some enhanced design features related to mitigating strategies that may be used by a potential COL applicant to satisfy the MBDBE rule requirements including enhancements to the 480 VAC RB 1E power centers.

GEH revised DCD Tier 1, Section 2.12.1, and Figure 2.12.1a, and DCD Tier 2, Table 1.9-1, Section 8.3.1.1.2.1, Figure 8.3-1, Sheet 3, and Section 8.3.4. The applicant's changes will add RB external connections to provide electrical power to the 480 VAC RB 1E power centers from an external power source.

During a public teleconference on September 7, 2017 (ADAMS Accession No. ML17311A055), GEH agreed to provide COL Information Items regarding the design enhancements related to off-site non-safety portable power. A portable power supply could be used during a beyond-design-basis event to supply site safety-related 480 VAC 1E power centers for an extended loss of alternating current (ac) power (usually referred to as an extended station blackout (SBO) condition). GEH documented the COL Information Items in a letter dated October 10, 2017, (ADAMS Accession No. ML17283A305).

GEH added two COL Information Items to the ABWR DCD. These COL Information Items would add actions for a future applicant to (1) describe in the FSAR the physical location of external connections for a portable diesel generator and (2) develop procedures for connecting the portable external diesel generators. These design changes were submitted in the applicant's January 23, 2017 letter, Enclosure 2 and in the letter dated October 10, 2017, which were incorporated in ABWR DCD, Revision 7.

*8.3.4.4.3 Technical Evaluation*

In its submittal dated January 23, 2017, the applicant provided ABWR design enhancements with RB external connections for providing electrical power to the 480 VAC RB 1E power centers. This change enhances the capability to provide electrical power to critical power centers from an external power source. GEH revised DCD Tier 1, Section 2.12.1, and Figure 2.12.1a; and DCD Tier 2, Table 1.9-1,

Section 8.3.1.1.2.1, Figure 8.3-1 sheet 3, and Section 8.3.4. DCD Tier 1, Section 2.12.1 identifies the DCD section that discusses the electrical power distribution system. DCD Tier 1, Figure 2.12.1a identifies the Class 1E electrical power distribution system. DCD Tier 2, Table 1.9-1 identifies the summary of ABWR standard plant COL Information Items. DCD Tier 2, Section 8.3.1.1.2.1 identifies the DCD section that discusses the Power Centers. DCD Tier 2, Section 8.3.4 discusses the COL License Information Items.

In the January 23, 2017 letter, Enclosure 2 DCD Tier 2, Section 8.3.1.1.2.1, the applicant added a new paragraph which states that, to cope with an extended loss of ac power

[13] The final MBDBE rule was published in the *Feder al Register* on August 9, 2019 (84 FR 39684) with an effective date of September 9, 2019.

(ELAP), external (to the RB) connections to each 1E RB divisional power center for portable external 480 VAC diesel generators are installed, normally isolated from the 480 VAC 1E divisional power centers by open 1E breakers.

The applicant also added two COL Information Items in DCD Tier 2, Section 8.3.4.5, “Physical Locations of Connections for Portable External Diesel Generators,” which states that the COL applicant will describe in the FSAR the details and physical locations of the connections for the portable external diesel generators and DCD Tier 2, Section 8.3.4.6, “Develop Procedures for Connecting Portable External Diesel Generators,” which states that the COL applicant will develop procedures for connecting the portable external diesel generators.

The staff reviewed the ABWR design changes to ensure that proper connections and isolation are maintained to minimize the probability of losing electric power from the onsite power supplies. In DCD Tier 2, Table 8.1-1, “Onsite Power System SRP Criteria Applicable Matrix”, and DCD Tier 2, Section 8.3.3.1, “Physical Separation and Independence,” state that the ABWR design conforms to RG 1.75 Revision 2. GDC 17 requires the independence of safety-related equipment and circuits, and auxiliary supporting features to be established and maintained via physical separation and electrical isolation, RG 1.75, Revision 2, also provides guidance on physical separation and electrical isolation. DCD Tier 2, Section 8.3.1.1.2.1, “Power Centers,” the low voltage Class 1E Power Distribution System and states that each 480V Class 1E bus in a division is physically and electrically independent of the other 480V buses in other divisions and non-Class 1E load groups. Since the external connections to each Class 1E RB divisional power center for portable external 480 VAC diesel generators are normally isolated from the 480 VAC 1E divisional power centers by open Class 1E breakers, the staff finds that physical separation and electrical isolation from the Class 1E system is maintained. Hence, the staff finds that the design conforms to the guidance in RG 1.75, Revision 2, as it relates to physical separation and electrical isolation and therefore, continues to be consistent with the requirements of GDC 17. Therefore, the staff finds the changes to be acceptable.

The applicant provided the necessary information in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant’s responses to Item No. 26, of the staff’s letter dated July 20, 2012. Therefore, Confirmatory Item 8.3.4.4-1 from the staff’s advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *8.3.4.4.4 Conclusion*

The staff reviewed the GEH ABWR design enhancement adding RB external connections for providing portable electrical power to the 480 VAC RB 1E power centers and the associated COL Information Items. This design continues to conform to the guidance in RG 1.75 Revision 2 and is therefore consistent with the requirements of GDC 17, as discussed above. Therefore, the changes are acceptable.

# 9 AUXILIARY SYSTEMS

## 9.1.1 New Fuel Storage

### *9.1.1.1 Regulatory Criteria*

In this section the staff reviews and evaluates the applicant's changes with regard to new fuel storage and handling for the GEH ABWR design in the ABWR DCD, Revision 7. A combined license (COL) applicant that references the GEH ABWR DC will incorporate the new fuel handling storage requirements and will implement the applicable ABWR procedures to address regulatory requirements for new fuel storage and handling.

In DCD Tier 2, Section 9.1, "Fuel Storage and Handling," approved as part of the ABWR DC rule in 1997 (10 CFR Part 52, Appendix A), onsite underwater storage of spent fuel assemblies and new fuel assemblies is provided by the spent fuel pool (SFP). The SFP fuel racks ensure that stored fuel is maintained in a suitable geometry to prevent criticality and provide cooling for all evaluated design conditions. In order to facilitate handling during fuel inspection and preparation, new fuel assemblies could also be safely stored as close as practicable to the spent-fuel storage pool work area, which is located in the new fuel storage vault (NFSV) in the reactor building.

In ABWR DCD, Revision 6, GEH proposed to revise the ABWR DCD to eliminate the use of the NFSV for the storage of new fuel assemblies. This change will result in the ABWR utilizing the SFP for the storage of new fuel prior to loading into the reactor. The SFP racks were previously evaluated by staff and found acceptable for storage of new fuel assemblies as part of the initial ABWR DC and, therefore, is not evaluated as part of the ABWR renewal review.

The applicant's proposal to remove the NFSV does not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), this design change is an "amendment," as this term is defined in Chapter 1 of this supplement and will correspondingly be evaluated using the regulations in effect at renewal.

The relevant requirements for this area of review and the associated acceptance criteria are in NUREG−0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 9.1.2, "New and Spent Fuel Storage," Revision 4, issued March 2007, as summarized below:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the ability of structures housing the facility and the facility itself to withstand the effects of natural phenomena such as earthquakes;
- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the structures housing the facility and the facility itself withstanding the effects of environmental conditions, externally-generated missiles, internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks so safety functions are not precluded;

- GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to the facility design for fuel storage and handling of radioactive materials;
- GDC 63, “Monitoring Fuel and Waste Storage,” as it relates to monitoring systems for detecting conditions that could cause the loss of decay heat removal capabilities for spent fuel assemblies, detecting excessive radiation levels, and initiating appropriate safety actions;
- 10 CFR § 20.1101(b) as it relates to keeping radiation doses as low as reasonably achievable (ALARA);
- 10 CFR § 50.68, “Criticality Accident Requirements,” as it relates to criticality monitoring or design to preclude criticality accidents; and
- 10 CFR § 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in accordance with the DC , the provisions of the Atomic Energy Act of 1954, as amended, and the NRC’s rules and regulations.

### *9.1.1.2 Summary of Technical Information*

ABWR DCD, Revision 5, was submitted as part of the GEH DC renewal application in December 2010. There is no difference between Revision 5 and Revision 4 of DCD Tier 2, Section 9.1 approved as part of the ABWR DC rule.

In ABWR DCD, Revision 6, the applicant proposed to eliminate the NFSV. The SFP will be utilized for storage of new fuel prior to loading into the reactor. This change generated a large number of conforming changes in DCD Tier 2, Section 9.1.

### *9.1.1.3 Technical Evaluation*

The ABWR design change in ABWR DCD, Revision 7, includes the revision of ABWR DCD Tier 2, Section 9.1, in order to remove references to the NFSV and the associated new fuel storage racks from the ABWR design.

The staff reviewed all the changes related to the removal of the NFSV and racks. The certified design already allowed for new fuel to be moved directly from receipt inspection to the SFP for storage before use in the reactor vessel. Therefore, in this FSER supplement, the staff did not review the capability of the SFP to store new fuel assemblies.

By eliminating the NFSV the applicant did not alter the new fuel handling path from receiving to loading in the vessel. In addition, the staff finds that this design change does not introduce a new potential accident to those previously evaluated and, therefore, does not impact the safety conclusion that the staff had previously reached in its FSER for the initially certified ABWR design as documented in NUREG–1503.

### *9.1.1.4 Conclusion*

Based on the evaluation provided in this supplement FSER, the staff concludes that the design change to remove the NFSV does not alter the staff safety findings in NUREG–1503, the staff FSER for the initially certified design. Therefore, the ABWR design, as modified, continues to meet all applicable regulatory requirements including GDC 2, GDC 4, GDC 61, GDC 63, 10 CFR § 20.1101(b), 10 CFR § 50.68, and 10 CFR § 52.47(b)(1) as reviewed by the staff in accordance with the associated SRP acceptance criteria in Section 9.1.2, Revision 4.

### *9.1.2.1 New and Spent Fuel Storage*

#### *9.1.2.1.1 Regulatory Criteria*

The originally certified GEH ABWR DCD describes the fuel racks in the spent fuel pool (SFP) as a seismic Category I structure, and states that the combined license (COL) applicant will perform the necessary confirmatory criticality and load drop analysis, including consideration of the free fall of a fuel assembly and its associated handling tool. In NUREG–1503, the NRC staff FSER approved the fuel storage racks in the SFP as described in the ABWR DCD, Revision 4, as part of the original ABWR DC. This is documented in Section 9.1.2, "Spent Fuel Storage," of NUREG–1503.

In the July 20, 2012 letter, the NRC staff identified 28 items for consideration by GEH as part of its application to renew the ABWR DC. In Item Nos. 19 and 20 of the letter, the applicant was requested to provide thermal-hydraulic analysis and criticality analyses of new and spent fuel racks.

In response to the staff, in a letter dated August 11, 2015 (ADAMS Accession No. ML15223B138), the applicant stated that the fuel racks are highly dependent on the specific rack design; therefore, these analyses are more appropriately addressed as a COL item, so GEH submitted changes to the COL license information regarding SFP thermal-hydraulic and criticality analysis. In this letter, the applicant also proposed to remove the new fuel storage vault from the ABWR design and instead use the racks in the SFP for storage of new fuel prior to loading into the reactor; this change is evaluated in Section 9.1.1, "New Fuel Storage," of this supplemental FSER.

The ABWR DCD, Revision 6, submitted on February 19, 2016, reflects the changes described above. Specifically, the applicant made changes to a thermal-hydraulic analysis COL Information Item in the DCD Tier 2, Section 9.1.6.8, "Spent Fuel Racks Thermal-Hydraulic Analysis," and additional criticality analysis information to the COL Information Item in DCD Tier 2, Section 9.1.6.3, "Spent Fuel Storage Racks Criticality Analysis."

Because the applicant proposed to provide clarifications consistent with the original understanding of the design information regarding SFP thermal-hydraulic and criticality analysis, it is a "modification," as this term is defined in Chapter 1 of this FSER supplement. Therefore, the staff evaluated this change using the regulations in effect at the time the certification was originally issued.

The relevant requirements for this area of review and the associated acceptance criteria are in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 9.1.1, Revision 2, "Criticality

Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, Revision 3, "New and Spent Fuel Storage," both issued July 1981, as summarized below:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criterion for Nuclear Power Plants," (GDC) 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the facility design for fuel storage, specifically item 4 of GDC 61, requiring the system to be designed with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal;
- GDC 62, "Prevention of Criticality in Fuel Storage and Handling," as it relates to the prevention of inadvertent criticality in the fuel storage system by physical systems or processes, preferably by use of geometrically safe configurations.

*9.1.2.1.2 Summary of Technical Information*

The changes proposed by GEH in the letter dated August 11, 2015, to address Items Nos. 19 and 20 of the staff's July 20, 2012, letter include both Tier 1 and Tier 2 changes to the ABWR DCD. A DCD markup based on Revision 5 of the ABWR DCD was provided in Enclosure 2 of the August 11, 2015, letter (ADAMS Accession No. ML15223B141). The changes augment a COL Information Item for the combined new and spent fuel storage racks in the SFP.

In Item No. 19 the staff requested that a thermal-hydraulic analysis be provided to evaluate the rate of naturally circulated flow and the maximum rack water exit temperatures. GEH stated that, because the thermal-hydraulic analysis of the fuel racks is highly dependent on the specific rack design, this item is more appropriately addressed as a COL item. DCD Tier 2, Section 9.1.6.8 already included a COL Information Item to provide a confirmatory thermal-hydraulic analysis to the NRC for the spent fuel racks that evaluates the rate of naturally circulated flow and the maximum rack water exit temperature. The ABWR DCD markup from the August 11, 2015, letter adds a reference to existing inspection, tests, analysis, and acceptance criterion (ITAAC) 2.5.6.4 and provides specific acceptance criteria, including that the analysis will use maximum decay heat generation rates for the worst-case power history. Also, the natural circulation flow through the rack arrangement should prevent water temperatures from exceeding 100 degrees Celsius (°C)/(212 degrees Fahrenheit (°F)) under normal, abnormal, and accident conditions.

In Item No. 20 the staff requested that a criticality analysis be provided. The certified ABWR DCD already contained a separate COL Information Item 9.1.6.3 for the spent fuel storage rack criticality analysis. The ABWR DCD markup from the August 11, 2015, applicant letter revised the COL Information Item in Section 9.1.6.3 to add a specific reference to existing DCD ITAAC 2.5.6.1, 2.5.6.2, and 2.5.6.3 and specific acceptance criteria and analysis assumptions. The applicant subsequently incorporated the changes described above in DCD Tier 2, Revision 6 which are also reflected in the most current DCD Revision 7.

*9.1.2.1.3 Technical Evaluation*

The staff reviewed the changes to the ABWR DCD to address Items Nos.19 and 20 of the NRC staff's letter dated July 20, 2012, to determine compliance with GDC 61 and GDC 62 related to stored fuel cooling and criticality accident requirements. The staff relied on the guidance originally used for the ABWR DC, including SRP Section 9.1.1, Revision 2, and SRP Section 9.1.2, Revision 3, for the review. The review considered the placement of new fuel in the SFP due to the removal of the new fuel storage vault and new fuel storage racks from the ABWR design and how this change affects the staff's original ABWR FSER for the certified design.

The staff reviewed the design criteria, design bases, and safety classification for the fuel storage racks and the provisions necessary to maintain a subcritical array and adequate natural circulation cooling. The staff concluded that the design changes and related commitments conform to the regulations applicable and in effect at the time of the original certification and do not alter the original staff FSER conclusions, as described in NUREG–1503, and which are summarized below.

GDC 61 requires that the fuel storage system be designed for adequate safety under normal and postulated accident conditions. As relevant here, the design must be capable of adequately cooling the stored fuel under normal and postulated accident conditions. Since the detailed rack design is not specified in the ABWR DCD, and will be determined by the supplier, COL Information Item 9.1.6.8 is used to specify the acceptance criteria for thermal-hydraulic analysis. The GEH 100°C (212°F) limit for natural circulation flow through the racks under normal, abnormal, and accident conditions will ensure that boiling is prevented, and that adequate cooling can be maintained. A confirmatory analysis will be performed by the COL applicant which considers the number of racks in the storage pool and the limiting decay heat loading under normal, abnormal, and accident conditions.

GDC 62 requires the prevention of criticality in the fuel storage system through the use of physical systems or processes, with preference given to the application of geometrically safe configurations. The applicant revised COL Information Item 9.1.6.3 to specify acceptance criteria for the criticality analysis. A confirmatory analysis will be performed by the COL applicant which considers the number of racks in the storage pool, fuel capacity, rack material, neutron poison content, and fuel center-to-center distance. The analysis must demonstrate that the storage racks can be maintained subcritical (i.e., $k_{eff} \leq 0.95$) when fully loaded.

The staff evaluated the applicant changes to COL Information Items 9.1.6.3 and 9.1.6.8 and determined that these changes do not alter the scope, or the staff FSER conclusion reached as part of the original ABWR DC as documented in NUREG–1503. GEH provided sufficient additional details to the COL Information Items for the fuel racks related to thermal-hydraulic and criticality analyses to ensure that the detailed rack design will meet the applicable regulations. Therefore, the staff finds the changes to COL Information Items 9.1.6.3 and 9.1.6.8 acceptable.

The staff confirmed that the changes were appropriately incorporated in the ABWR DCD Revision 6, which are also reflected in the most current ABWR DCD Revision 7.

#### *9.1.2.1.4 Conclusions*

The staff reviewed the applicant's changes to the ABWR DCD, Revision 7, as described above. Based on this evaluation, the staff concludes that the revisions to the COL Information Items as described meet all applicable regulatory requirements at the time of original certification, specifically GDC 61 and 62, and therefore these COL Information Item clarifications as reflected in the ABWR DCD, Revision 7, are acceptable.

### *9.1.2.2 Fuel Racks*

#### *9.1.2.2.1 Regulatory Criteria*

In the ABWR DCD, Revision 7, the applicant provided changes to the accident load combinations and the fuel rack support description, along with changes to two combined license (COL) license information items.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 18 Part B of the July 20, staff letter, the applicant was requested to provide structural, dynamic, and impact analysis of new and spent fuel racks in DCD Tier 2, Section 9.1.6.2, "Dynamic and Impact Analysis of New Fuel Storage Racks," and DCD Tier 2, Section 9.1.6.7, "Spent Fuel Racks Structural Evaluation."

The originally certified ABWR DCD, Revision 4, identifies that the fuel racks in the spent fuel pool (SFP) are seismic Category I structures. The staff evaluation documented in NUREG–1503, Section 9.1.2, "Spent Fuel Storage," approved the fuel storage racks in the SFP as described in the ABWR DCD.

In a letter dated August 11, 2015 (ADAMS Accession No. ML15223B139) GEH submitted proposed changes to the accident load combinations and fuel racks support description, along with a revised COL license information item in DCD Tier 2, Section 9.1.6.7. These changes have the effect of deferring the structural, dynamic, and impact analysis of the spent fuel racks to the COL applicant. In addition, the applicant deleted the COL license information item related to the dynamic and impact analysis of the new fuel storage racks described in DCD Tier 2, Section 9.1.6.2, Revision 5, as part of the removal of the new fuel storage vault evaluated by the staff in Section 9.1.1, "New Fuel Storage," of this supplemental FSER.

Because a potential COL applicant will perform the pertinent spent fuel rack analyses in accordance with the regulations in effect during the COL application review, these changes are "amendments," as defined in Chapter 1 of this supplement. Therefore, in accordance with 10 CFR § 52.59(c), these design analysis changes will be evaluated by the staff using the regulations in effect at renewal.

The relevant requirements of the NRC's regulations for this area of review, and the associated acceptance criteria, are in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 9.1.2, "New and Spent Fuel Storage," Revision 4, issued March 2007, and Appendix D to SRP Section 3.8.4, "Other Seismic Category I Structures," Revision 4, issued September 2013. The applicable requirement for this review is:

- 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," requires that structures housing the facility and the facility itself can withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and appropriate combinations of all loads.

### 9.1.2.2.2 *Summary of Technical Information*

GEH submitted the markups to the ABWR DCD, Revision 5, in Enclosure 2 of the applicant letter dated August 11, 2015. In Enclosure 1 (ADAMS Accession No. ML15223B140) of this letter, GEH stated that structural, dynamic, and impact analyses of the fuel racks are more appropriately addressed as a COL information item since these analyses are highly dependent on the specific rack design. The staff confirmed that the ABWR DCD, Revision 5, markups were included in the ABWR DCD, Revision 6, and is reflected in the most current ABWR DCD Revision 7.

### 9.1.2.2.3 *Technical Evaluation*

The staff reviewed the changes in DCD Tier 2, Section 9.1.2.1.3, "Mechanical and Structural Design," DCD Tier 2, Section 9.1.2.3.2, "Structural Design and Material Compatibility Requirements," and DCD Tier 2, Section 9.1.6.7, "Spent Fuel Racks Structural Evaluation," to ensure that the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes on the fuel racks in the ABWR design are considered, as required by GDC 2, using the guidance in SRP Section 9.1.2, Revision 4, and Appendix D to SRP Section 3.8.4, Revision 4.

The ABWR DCD, Revision 5, did not provide the structural, dynamic, and impact analyses of the fuel racks. In its August 11, 2015 letter, GEH addressed these analyses through a COL information item and provided additional details regarding the structural, dynamic, and impact analyses of the fuel racks for the SFP. The applicant did not address the new fuel racks because the applicant proposes to remove these racks and store new fuel in the SFP; this change is evaluated in Section 9.1.1, of this supplemental SER. Therefore, only the structural, dynamic, and impact analyses of fuel racks in the SFP were evaluated as part of this ABWR supplemental FSER section.

In DCD Tier 2, Section 9.1.2.1.3, GEH replaced the list of load combinations previously specified with a reference to the load combinations in Appendix D to SRP Section 3.8.4. The staff finds these revisions acceptable because GEH followed the guidance of Appendix D to SRP Section 3.8.4 Revision 4. GEH also deleted language regarding the use of linear elastic design methods in the structural evaluation of the fuel racks. This deletion is also acceptable because (a) the structural, dynamic, and impact analyses of fuel racks is a COL license information item, (b) design methods can be determined by the COL applicant, and (c) the localized nonlinear plastic regime may occur due to postulated loading cases as proposed by a future COL applicant. In addition, GEH changed the text from "the dynamic method" to "an acceptable dynamic analysis method," which is acceptable because the COL applicant will identify the dynamic analysis method, and the staff will determine the acceptability of the method during the COL application review. GEH deleted the statement "Compressive stability will be calculated according to the American Iron and Steel Institute (AISI) code for light gauge structures," which is also acceptable to the staff because light gauge structures are no longer used as part of fuel rack fabrication and the AISI code is not referenced for acceptable fuel rack design in Appendix D to SRP Section 3.8.4, Revision 4.

In DCD Tier 2, Section 9.1.2.3.2, GEH revised the sentence related to an older SFP rack design that utilized a sub-structure with the description of updated designs that are considered "Freestanding" as follows:

> The fuel storage racks are designed to be supported vertically by the fuel floor. The support structure allows sufficient pool water flow for natural convection cooling of the stored fuel. The fuel rack modules are freestanding (i.e., not attached to the floor and can be removed).

The staff reviewed the design change and found it acceptable because GEH accurately described the boundary conditions of the SFP fuel racks. In addition, GEH deleted the statement "Lead-in guides at the top of the storage spaces provide guidance of the fuel during insert," which is acceptable to the staff because it is not related to the structural design and material compatibility requirements of the fuel racks.

In DCD Tier 2, Revision 6, Section 9.1.6.7, GEH added the following italicized language to a COL Information Item:

> The COL applicant shall provide the NRC a confirmatory structural evaluation of the spent fuel racks, as outlined in Subsection 9.1.2.1.3. *This evaluation is dependent on a vendor specific design and the as-built configuration of spent fuel storage racks.*
> *Structural integrity of the racks will be demonstrated for the load combinations described in SRP 3.8.4 Appendix D. The fuel storage racks meet seismic Category I requirements.*

The staff reviewed the changes and found them acceptable because GEH provided additional details related to the COL license information for the fuel racks structural evaluation, and GEH refers to the guidance of Appendix D to SRP Section 3.8.4, Revision 4.

##### *9.1.2.2.4 Conclusion*

The staff's review concludes that the applicant's changes related to the fuel racks for GEH ABWR design comply with GDC 2, and the COL applicant will provide the detailed structural evaluations of the fuel racks in accordance with the guidance of Appendix D to SRP Section 3.8.4, Revision 4, which is acceptable.

### 9.1.3 Fuel Pool Cooling and Cleanup System

In a letter dated August 25, 2015 (ADAMS Accession No. ML15237A192), the applicant added spent fuel pool (SFP) level instruments that conform with applicable guidance specified in the Japan Lesson-Learned Project Directorate-Interim Staff Guidance (JLD-ISG)-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," dated August 29, 2012 (ADAMS Accession No. ML12221A339). That guidance endorses with exceptions and clarifications, the methodologies described in the Nuclear Energy Institute (NEI) industry guidance document NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," issued August 2012 (ADAMS Accession No. ML122400399).

This change to the design of SFP instruments resulted in changes to the DCD Tier 2, Section 9.1.3, "Fuel Pool Cooling and Cleanup System," which incorporated safety-related SFP instrumentation consisting of two independent wide-range level transmitters that transmit water level signals to the main control room. In addition, the water level signals will also be provided to the remote shutdown panels or other appropriate location accessible post-accident.

In addition, this change resulted in changes to the following ABWR DCD sections:

- DCD Tier 1, Section 2.6.2, “Fuel Pool Cooling and Cleanup System” including Figure 2.6.2 and Table 2.6.2
- DCD Tier 2, Chapter 1, Tables 1.8-21 and 1.8-22These ABWR design enhancements would provide a potential COL applicant the means for meeting the rule requirements of 10 CFR § 50.155, “Mitigation of Beyond-Design-Basis Events,” (MBDBE rule) regarding requirements for safety-related SFP Instrumentation which codified the requirements stemming from Commission Order EA-12-051. The final MBDBE rule was published in the Federal Register on August 9, 2019 (84 FR 39684) with an effective date of September 9, 2019.

The staff review of these changes and other changes associated with the new SFP instrumentation is provided in Section 22.2 of this FSER supplement.

### 9.1.4 Light Load Handling System (Related to Refueling)

#### *9.1.4.1 Regulatory Criteria*

In this supplemental FSER section the staff reviewed and evaluated the ABWR DC renewal applicant’s changes to the light load-handling system (LLHS) for the GEH ABWR design. The LLHS provides the means of transporting, handling, and storing fuel (both new and spent fuel) in the reactor building.

A combined license (COL) applicant that references the renewed ABWR DC will incorporate the ABWR LLHS as specified by the ABWR DCD, Revision 7, for the safe handling of new and spent fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling.

In ABWR DCD, Revision 6, GEH revised the ABWR DCD to eliminate the use of the new fuel storage vault (NFSV) and its new fuel storage racks. This design change will result in the ABWR utilizing the spent fuel pool (SFP) for storage of new fuel prior to loading into the reactor. The change to the DCD LLHS to remove the NFSV does not fall within the definition of a “modification.” Therefore, in accordance with 10 CFR § 52.59(c), this design change is an “amendment,” as this term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations in effect at renewal.

The relevant requirements for this area of review and the associated acceptance criteria are given in NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) Section 9.1.4, “Light Load Handling System and Refueling Cavity Design,” Revision 4, issued July 2014, as summarized below:

- 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” (GDC) 2, “Design

Bases for Protection Against Natural Phenomena," as it relates to structures housing the system, and the system itself, being capable of withstanding the effects of earthquakes;

- GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to radioactivity release as a result of fuel damage, and the avoidance of excessive personnel radiation exposure;
- GDC 62, "Prevention of Criticality in Fuel Storage and Handling," as it relates to prevention of inadvertent criticality;
- 10 CFR § 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's rules and regulations.

*9.1.4.2 Summary of Technical Information*

GEH submitted the ABWR DCD, Revision 5, as part of its DC renewal application in 2010. There was no difference between ABWR DCD Revision 5 and Revision 4 of DCD Tier 2, Section 9.1.4, approved as part of the ABWR DC rule in 1997 (10 CFR Part 52, Appendix A).

In ABWR DCD, Revision 6, submitted by the applicant in 2016, GEH proposed to eliminate the NFSV. Therefore, the racks in the SPF will be utilized for storage of new fuel prior to loading into the reactor. The elimination of the NFSV was evaluated by the staff in Section 9.1.1, "New Fuel Storage," of this supplemental FSER. This design change generated conforming changes in the following ABWR DCD Sections:

- DCD Tier 1, Section 2.15.3 "Cranes and Hoists", is revised to eliminate references to the new fuel storage vault and references to dry storage of new fuel, which was to be done in the new fuel storage vault.
- DCD Tier 2, Section 9.1.4 "Light Load Handling System (Related to Refueling)," the applicant revised the process of receiving and handling of new fuel assemblies to eliminate any step that stores new fuel into the new fuel vault or make reference to the new fuel racks or new fuel storage vault.

*9.1.4.3 Technical Evaluation*

The staff evaluated the ABWR LLHS related to the ABWR SFP as part of the initially certified ABWR DC and it was found acceptable for handling new and spent fuel assemblies. Therefore, the staff did not re-evaluate the ABWR LLHS as part of the ABWR renewal review.

The ABWR DCD design changes include the revision of DCD Tier 1, Section 2.15.3 and DCD Tier 2, Section 9.1.4, in order to remove references to the NFSV and its associated storage racks, and to clearly indicate that the SFP is the only storage location for new fuel assemblies.

By eliminating the NFSV, the applicant has not altered the new fuel transportation path, previously reviewed as part of the initial ABWR DC, from receiving to loading new fuel in the SFP. The original design included the option to put new fuel in the NFSV prior to moving it to the SFP, but the applicant proposed to eliminate this option as part of its ABWR DCD, Revision 6, submittal. The staff finds that this change does not introduce a new accident scenario to those previously evaluated, and it does not impact the safety conclusion that the staff has previously reached in the FSER for the initially certified design as documented in NUREG–1503.

#### *9.1.1.4 Conclusion*

Based on the evaluation provided in this section of the FSER supplement for the ABWR DC renewal, the staff concludes that the design change to remove the NFSV and the change to the LLHS related to new and spent fuel handling as documented in the ABWR DCD, Revision 6 and as reflected in the ABWR DCD, Revision 7, to address the elimination of the option to use the NFSV does not alter the staff's safety findings in NUREG–1503. Therefore, this ABWR design change meets all applicable regulatory requirements in GDC 2, GDC 61, GDC 62, and 10 CFR § 52.47(b)(1), as reviewed by the staff in accordance with the associated acceptance criteria in NUREG–800, Section 9.1.4, Revision 4.

### 9.1.5 Overhead Heavy Load Handling Systems

#### *9.1.5.1 Regulatory Criteria*

In this section the staff reviews and evaluates the applicant's proposed changes to the ABWR DCD, Revision 7, overhead heavy load handling system (OHLHS), which consists of all components and equipment for moving all heavy loads. This includes loads weighing more than one fuel assembly and its handling device - loads greater than 1,000 pounds for the GEH ABWR design. The main emphasis in the ABWR DC renewal review is on critical load handling where inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, a criticality accident, or an inability to cool fuel within the reactor vessel or spent fuel pool; or could prevent the safe shutdown of the reactor.

A combined license (COL) applicant that references the GEH ABWR DC will incorporate the OHLHS requirements specified for the ABWR design and the COL applicant will implement the applicable ABWR procedures to address regulatory requirements for overhead heavy load handling as described in the ABWR DCD.

GEH submitted ABWR DCD, Revision 5, as part of the GEH DC renewal application in 2010. There was no difference between Revision 5 and Revision 4 of DCD Tier 2, Section 9.1.5, approved as part of the original ABWR DC in 1997. In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. The applicant was requested in Item No. 13, of that letter, to consider adding a commitment to American Society of Mechanical Engineers (ASME) standard NOG-1, "Rules for Construction of Overhead and Gantry Cranes," issued in 2004, as an acceptable approach to meeting the criteria in NUREG–0554, "Single Failure-Proof Cranes for Nuclear Power Plants," issued May 1979, for the design of OHLHS cranes.

The applicant had proposed changes in ABWR DCD, Revision 6, submitted in 2016, to update the DC renewal application and to identify ASME NOG-1 as an acceptable approach for a COL applicant to design the OHLHS and meet the design requirements already established for the ABWR DC. The applicant also made changes to reflect the elimination of the new fuel storage vault, as discussed in Section 9.1.1 of this FSER supplement. These changes do not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), these design changes are "amendments," as this term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations in effect at renewal.

The relevant requirements for this area of review and the associated acceptance criteria are given in NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 9.1.5, Revision 1, "Overhead Heavy Load Handling Systems," issued March 2007, Review criteria are as follows:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 1, "Quality Standards and Records," as it relates to the design, fabrication, and testing of structures, systems, and components (SSCs) important to safety 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 the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes.
- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the protection of safety-related equipment from the effects of internally generated missiles (i.e., dropped loads).
- Acceptance criteria adequate to meet the above requirements are:
- Acceptance for meeting the relevant aspects of GDC 1, GDC 2, and GDC 4 for heavy load handling cranes is based on NUREG–0554; and
- ASME NOG-1 is one acceptable approach to meet the requirements of NUREG-0554.

*9.1.5.2 Summary of Technical Information*

In Revision 6 of the ABWR DCD, the applicant proposed the following changes related to the OHLHS:

- DCD Tier 2, Section 9.1.5.1, "Design Bases," is revised, in part, to state as indicated below (added texts are underlined):

  " … Cranes and hoists are also designed to criteria and guidelines of NUREG–0612, Subsection 5.1.1(7), ANSI B30.2 and CMAA-70 specifications for electrical overhead traveling cranes, including ANSI B30.11, ANSI B30.16, and NUREG–0554 as applicable. <u>For design of Type 1 cranes, ASME NOG-1 is an acceptable approach to meeting NUREG–0554 criteria</u>."

- DCD Tier 2, Section 9.1.5.2.1, “Reactor Building Crane,” is revised, in part, to state as indicated below (added texts are underlined and deleted texts are crossed-out):

  “ … The main hook 1.471 MN will be used to lift the concrete shield blocks, drywell head, reactor pressure vessel (RPV) head insulation, RPV head, dryer, separator strongback, RPV head strongback carousel, new-fuel shipping containers, and spent-fuel shipping cask. The orderly placement and movement paths of these components by the R/B crane precludes transport of these heavy loads over the spent fuel storage pool~~ or over the new-fuel storage vault~~.

  … Minimum crane coverage includes R/B refueling floor laydown areas, and R/B equipment storage pit. During normal plant operation, the crane will be used to handle new-fuel shipping containers and the spent-fuel shipping casks. Minimum crane coverage must include~~ the new-fuel vault,~~ the R/B equipment hatches, and the spent-fuel cask loading and washdown pits. A description of the refueling procedure can be found in Section 9.1.4.

  The R/B crane will be interlocked to prevent movement of heavy loads over the spent-fuel storage portion of the spent-fuel storage pool. Since the crane is used for handling large heavy objects over the open reactor, the crane is of Type I design. The R/B crane shall be designed to meet the single-failure-proof requirements of NUREG–0554. For design of Type 1 cranes, ASME NOG-1 is an acceptable approach to meeting NUREG–0554 criteria.”

- DCD Tier 2, Table 9.1-6, “Reference Codes and Standards,” is revised to add a new entry as indicated below:

  “ASME NOG-1    Rules for Construction of Overhead and Gantry Cranes (Top Running Bridges, Multiple Girder)”

- DCD Tier 2, Table 9.1-8, “Heavy Load Operation,” under the column of “**Hardware Handling Tasks**,” is revised, in part, to state as indicated below (added text are underlined and deleted texts are crossed-out):

  “ … Remove inner container and ~~store fuel bundle in new fuel vault rack. M~~move fuel to new fuel inspection stand~~, inspect and return to storage~~ and perform inspection.

  Move new fuel from~~ vault~~ inspection stand to fuel pool, storage of fuel channel fixtures. Channel new fuel and store. Move channeled fuel and load into reactor core … ”

- DCD Tier 2, Table 1.8-19, “Standard Review Plans and Branch Technical Positions Applicable to ABWR,” is revised to update SRP 9.1.5 from “Revision 0 (issued July 1981)” to “Revision 1 (issued March 2007).”

- DCD Tier 2, Table 1.8-21, “Industrial Codes and Standards* Applicable to ABWR,” is revised to add a new entry under the heading of “**American Society of Mechanical Engineers (ASME)”** as indicated below:

  “NOG-1 2004  Rules for Construction of Overhead and Gantry Cranes”

### *9.1.5.3 Technical Evaluation*

The staff reviewed all changes to the OHLHS in the ABWR DCD Revision 6 in accordance with SRP Section 9.1.5.

The ABWR DCD originally referenced NUREG–0554 as the established guidance for the design of the reactor building (R/B) crane. During the staff's review of Revision 4 of the ABWR DCD, referencing NUREG–0554 alone was sufficient to meet acceptance criteria delineated in Revision 0 (issued July 1981) of SRP Section 5. However, in Revision 1 (issued March 2007) of SRP Section 9.1.5, the staff enhanced the guidelines for the design of single-failure-proof cranes by adding the ASME NOG-1-2004 standard, "Rules for Construction of Overhead and Gantry Cranes," which provides comprehensive detailed design requirements including information that shows how specific design criteria of NUREG–0554 will be satisfied. In Revision 6 of the ABWR DCD, the applicant added a reference to ASME NOG-1, 2004 as an acceptable approach to meeting NUREG–0554 criteria for the design of the R/B crane. The staff finds the change acceptable.

The staff also noted that the changes to DCD Tier 2, Section 9.1.5.2.1 and Table 9.1-8 include deletion of "the new-fuel storage vault" from various descriptions of load handling activities involving the use of the R/B crane. In the ABWR DCD, Revision 6, the applicant proposes a design change which removes the new fuel storage facility from the scope of the ABWR DC. The new fuel, upon receipt at the site, will be stored instead in the spent fuel pool as described in the applicant's changes to DCD Sections 9.1.1, "New Fuel Storage." The staff's evaluation of this design change is documented in Section 9.1.1 of this FSER supplement. As such, the proposed changes to DCD Tier 2, Section 9.1.5 and Table 9.1-8 are only conforming changes; therefore, the staff finds them acceptable.

### *9.1.5.4 Conclusion*

Based on the evaluation provided in this FSER supplement, the staff concludes that the amendment as reflected in ABWR DCD, Revision 7, associated with the addition of ASME NOG-1 as an option for designing the cranes meets the requirements of a GDC 1, GDC 2, and GDC 4 as reviewed by the staff in accordance with the associated SRP Section 9.1.5, Revision 1, acceptance criteria. The staff also concludes that the conforming changes to DCD Tier 2, Section 9.1.5.2.1 and Table 9.1-8, to reflect the elimination of the new fuel storage vault are acceptable.

## 9.5.1 Fire Protection System

### *9.5.1.1 Regulatory Criteria*

In the ABWR DCD, Revision 7, the applicant included changes to DCD Tier 2, Section 9.5.1, "Fire Protection System," that were submitted as part of ABWR DCD, Revision 6, and associated DCD markups, which clarify the likelihood of multiple spurious actuations (also called "multiple spurious operations") due to fire in digital systems. The applicant also included changes to the ABWR DCD to require combined license (COL) applicants to follow the methodology in Nuclear Energy Institute (NEI) 00-01, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Revision 2, issued June 2009, as modified by Regulatory Guide (RG) 1.189, "Fire Protection for Nuclear

Power Plants," Revision 2, issued October 2009, to address multiple spurious actuations in analog systems. These changes are limited to clarifying the language in the ABWR DCD in regard to the likelihood of multiple spurious actuations due to a fire, clarifying the description of the defense-in-depth architecture of the digital systems that would prevent a spurious signal from becoming a spurious actuation, and specifying the methodologies to be used by COL applicants when addressing multiple spurious actuations in analog systems for compliance with 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 3, "Fire Protection," and compliance with 10 CFR § 50.48, "Fire protection," as these regulations existed in 1997. The changes are ABWR DCD clarifications of the existing design and therefore, they are "modifications," as that term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations applicable and in effect at the initial ABWR certification.

The following regulatory requirements are relevant for this area of review:

- 10 CFR § 50.48 (1997), "Fire protection," subsection (a) which requires, in part, a description of "the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured."
- GDC 3 (1997), "Fire Protection," as it relates to the fire protection program of the GEH ABWR standard plant design.

The staff conducted its review in accordance with NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 9.5.1, "Fire Protection Program," Revision 3, issued July 1981. In addition, the staff's review followed the guidance in RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, with respect to multiple spurious actuations in analog systems.

### *9.5.1.2 Summary of Technical Information*

In NUREG–1503, Section 9.5.1, "Fire Protection System," the staff FSER for the originally certified ABWR DC, did not evaluate digital instrumentation & control (I&C) systems response and spurious actuations given a fire event. In the originally certified ABWR in DCD Tier 2, Section 3.13.4.2, "Fire Events," it stated the following:

> "… [S]eparation criteria are maintained during design basis fire events. Internal fire in one affected zone will not propagate to other [redundant] divisions. Smoke is removed from the affected zone. Other zones are pressurized and also vented."

Therefore, the ABWR is designed to maintain safe shutdown capabilities following a fire in any affected zone. In addition, in DCD Tier 2, Section 9.5.1.1.7, "Spurious Control Actions," the originally certified design DCD stated, "The probability of two spurious signals matching is essentially zero."

In RAI 09.05.01-1, dated April 29, 2015 (ADAMS Accession No. ML15118A725), the staff requested that GEH perform an evaluation for the effects of multiple spurious operations due to a fire consistent with NEI 00-01, Revision 2, as modified in RG 1.189, Revision 2 or to propose and justify an alternative.

### *9.5.1.3 Technical Evaluation*

The applicant provided responses to the staff's RAI in letters dated July 30, 2015 (ADAMS Accession No. ML15212A762), October 29, 2015 (ADAMS Accession No. ML15302A308), April 11, 2016 (ADAMS Accession No. ML16102A344), and December 7, 2016 (ADAMS Accession No. ML16342C331), including proposed ABWR DCD markups. GEH stated that a detailed assessment of the ABWR's vulnerability to multiple spurious operations would need to be conducted during the detailed design phase. GEH provided the following changes to DCD Tier 2, Section 9.5.1.1.7 and DCD Tier 2, Section 9.5.13.22:

> a requirement that the COL applicant provide an evaluation of the ABWR's susceptibility to Multiple Spurious Operations (MSO) in accordance with the methodology contained in NEI 00-01, Revision 2, and as modified by Regulatory Guide 1.189, Revision 2. The COL applicant will submit the results of this evaluation to the NRC for review.

The staff finds this acceptable since RG 1.189, Revision 2, endorses NEI 00-01, Revision 2. The ABWR DCD now provides an acceptable methodology for performing the fire-induced multiple spurious analysis, whereas the original ABWR DC did not specify a methodology.

The applicant also addressed multiple spurious actuations due to fire in digital systems by proposing several changes to DCD Tier 2, Section 9.5.1.1.7. First, the applicant proposed to replace the words "probability …is essentially zero" with "likelihood … is miniscule." The staff finds this acceptable because the revised ABWR DCD no longer implies a probabilistic analysis was used, which is consistent with the application of a deterministic fire protection program. In addition, the applicant proposed to insert language to clarify that along with optical fiber cabling, fire-induced spurious actuation will be considered in main control room components, remote multiplexing units (RMU), essential multiplexing system (EMS) and digital controller equipment in the control building connected via fiber-optic cable. The staff finds this acceptable because this change properly expands the spurious actuation analysis to include the digital equipment both in and outside of the main control room fire area. Lastly, the applicant proposed to insert language describing the defense-in-depth of the digital architecture that would prevent a spurious signal from becoming a spurious actuation. The digital architecture utilizes message authentication which requires the message format and sequence to be correct and to be recognized. The staff finds this acceptable because it makes use of features that are pertinent to digital systems.

The staff finds acceptable the changes described above because they clarify the language in the ABWR DCD in regard to the likelihood of multiple spurious actuations due to a fire and the description of the defense-in-depth of the digital architecture that would prevent a spurious signal from becoming a spurious actuation. In addition, the ABWR DCD specifies the NRC approved methodologies that COL applicants would use when addressing multiple spurious actuations in analog systems.

The applicant provided the necessary information in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response to 09.05.01-1. Therefore, Confirmatory Item 9.5.1-1 from the staff's advanced safety evaluation report with no open items for the ABWR DC renewal is resolved and closed.

### *9.5.1.4 Conclusion*

Based on the evaluation provided in this supplemental FSER section, the staff concludes that the changes in the ABWR DCD, Revision 7, do not alter the safety findings made in the FSER for the original ABWR DC. In addition, the changes by the applicant to address multiple spurious actuations in analog systems, are in accordance with updated guidance in RG 1.189, Revision 2. Therefore, the staff finds that the changes comply with 10 CFR § 50.48 (1997) and GDC 3 (1997) and are therefore acceptable.

# 11 RADIOACTIVE WASTE MANAGEMENT

## 11.4 Solid Waste Management System

Section 11.4.2, of NUREG–1503, the staff FSER for the original ABWR DC, indicates that the solid waste management system meets the guidance of Regulatory Guide (RG) 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Revision 1, issued October 1979. However, in reviewing DCD Tier 2, Section 11.4 for the GEH DC renewal application, it was unclear that the solid waste management system was to be designed, constructed, and tested in accordance with the criteria in RG 1.143. In addition, there were apparent discrepancies in the ABWR DCD regarding the off-gas system and off-gas vault design (which are mostly discussed in DCD Tier 2, Section 11.3). This supplemental evaluation documents the staff’s review of the design of the radioactive waste management system, as it relates to conformance with RG 1.143 and compliance with the associated regulatory requirements in the ABWR DCD Revision 7.

### *11.4.1 Regulatory Criteria*

As explained below, the ABWR DCD changes related to the solid waste management system are to supply information omitted from the originally certified DCD to ensure that the solid waste management system meets the regulations applicable and in effect at initial certification. The changes related to the off-gas system and off-gas vault are to correct errors and inconsistencies in the originally certified ABWR DCD associated with the off-gas system and off-gas vault design descriptions to ensure that the off-gas system and off-gas vault meets the regulations applicable and in effect at initial certification. Therefore, the changes are “modifications,” as this term is defined in Chapter 1 of this supplement and will be evaluated using the regulations applicable and in effect at initial certification.

The following regulatory requirements provide the basis for the acceptance criteria for the staff’s review:

- 10 CFR § 52.47(a)(1)(i) (1997), requires that the DC application must contain the technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, “Standards for Protection against Radiation,”10 CFR Part 50,“Domestic Licensing of Production and Utilization Facilities,” and its appendices, and 10 CFR Part 73, “Physical Protection of Plants and Materials,” and 10 CFR Part 100, “Reactor Site Criteria,” that is technically relevant to the design and not site-specific.
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 2, “Design Bases for Protection Against Natural Phenomena” (1997), requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. The design bases for these structures, systems, and components must reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate

combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

- 10 CFR Part 50, Appendix A, GDC 60, “Control of Releases of Radioactive Materials to the Environment” (1997), requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes during normal reactor operation, including anticipated operational occurrences.
- 10 CFR Part 50, Appendix A, GDC 61, “Fuel Storage and Handling and Radioactivity Control” (1997), requires the fuel storage and handling, radioactive waste, and other systems that might contain radioactivity to be designed to assure adequate safety under normal and postulated accident conditions, including appropriate containment, confinement, and filtering systems for radioactive waste systems.

### *11.4.2 Summary of Technical Information*

RG 1.143, provides, in part, design, construction, and testing criteria for radioactive waste management structures, systems, and components at nuclear power plants. Following the guidance of RG 1.143 ensures that the radioactive waste management systems comply with the pertinent portions of the GDC within the scope of this supplement SER discussed above. GEH ABWR DCD Tier 2, Table 1.8-20, indicates that RG 1.143, Revision 1, is applicable to the ABWR design and DCD Tier 2, Sections 11.2 and 11.3, provide information and commitments that ensure systems and components of the liquid and gaseous waste management systems (including associated structures) are designed and tested in accordance with RG 1.143. However, while NUREG–1503, Section 11.4.2, indicates that the solid waste management system meets the guidelines of RG 1.143, there was no specific commitment in the DCD that the solid waste management system would be designed, constructed, and tested in accordance with RG 1.143. RG 1.143, Regulatory Position 3, specifies the design and testing criteria for solid waste management systems and Regulatory Position 6 provides the quality assurance criteria.

In RAI 11.04-1, dated March 12, 2015 (ADAMS Accession No. ML15069A674), the staff requested that GEH provide information ensuring that the solid waste management system conforms with RG 1.143, Revision 1, or provide an alternative approach to meeting the NRC regulations. In addition, while the ABWR DCD Tier 2, Section 11.3, specifies that the off-gas system is designed in accordance with RG 1.143, DCD Tier 2, Table 3.2-1, contained several apparent errors and inconsistencies that could potentially create confusion regarding the off-gas system and off-gas vault design. Therefore, the staff requested that the applicant also correct these errors and inconsistencies in ABWR DCD Tier 2, Table 3.2-1.

### *11.4.3 Technical Evaluation*

In its response to RAI 11.04-1 (ADAMS Accession No. ML15099A586), the applicant proposed updating DCD Tier 2, Section 11.4.1.2, to specify that the solid waste management system design “compli[es] with Regulatory Guide 1.143.” This would include any mobile equipment that is used. In addition, in the response, the applicant corrected the errors associated with the ABWR DCD Tier 2, Table 3.2-1, which clarifies that the off-gas system and off-gas vault will be designed in accordance with RG 1.143, Revision 1.

In Supplement 1 of its response to RAI 11.04-1 (ADAMS Accession No. ML15202A045), the applicant proposed including additional information in ABWR DCD Tier 2, Section 11.4.1.2, not only to specify that the solid waste management system complies with RG 1.143, but also to state that this includes the quality classification, construction, and testing requirements in DCD Tier 2, Section 11.2.1.2.1, and the building requirements in DCD Tier 2, Section 11.2.1.2.2. These sections provide the design information, including codes and standards, consistent with RG 1.143, Revision 1, for which the solid waste management system must be designed. Therefore, the response to RAI 11.04-1, including Supplement 1, provides DCD changes which ensure that the solid waste management system and off-gas system (including associated structures) are designed, constructed, and tested, in accordance with RG 1.143, Revision 1. The staff finds this to be acceptable.

The staff verified that the DCD changes described in the response to RAI 11.04-1, including Supplement 1, were incorporated into DCD Revision 7. Therefore, this issue is resolved.

### *11.4.4 Conclusion*

Based on the above, the ABWR DCD, Revision 7, meets the requirements of 10 CFR § 52.47(a)(1)(i) (1997). In addition, the design, construction, and testing criteria for the structures, systems, and components associated with the solid radioactive waste management system and off-gas system and off-gas vault conform to the guidance in RG 1.143, Revision 1, and the information in the DCD is now consistent regarding the design of the off-gas system and off-gas vault. Conformance with RG 1.143, Revision 1, in combination with other aspects of the design, including the design requirements of these structures, the control of radioactive effluents, the radiation shielding design, and other radiation protection design features ensure that these structures, systems, and components are in compliance with 10 CFR Part 50, Appendix A, GDC 2, 60, and 61. Therefore, these ABWR DC design changes are acceptable.

# 12 RADIATION PROTECTION

## 12.4 Radiation Sources

This supplemental FSER documents the staff's review of the applicant's ABWR DC renewal incorporation of the condensate storage tank (CST) as a radiation source and design features associated with reducing radiation exposure and minimizing potential contamination from the CST in DCD Tier 2, Section 12.2, "Radiation Sources." This supplement also contains updated information clarifying that the inspections, tests, analyses, and acceptance criteria (ITAAC) in DCD Tier 1, Tables 3.2a and 3.2b should have been identified as design acceptance criteria (DAC). Finally, this supplemental evaluation also documents the staff's review of source term table errors and associated corrections in DCD Tier 2, Section 12.2.

### *12.4.1 Regulatory Criteria*

As explained below, the CST design changes are to supply information omitted from the originally certified DCD that is necessary to meet the regulations applicable and in effect at initial certification. The clarification of the ITAAC in DCD Tier 1, Tables 3.2a and 3.2b as DAC is consistent with the original understanding of these ITAAC and is needed to correct an inconsistency with DCD Tier 2 of the originally certified ABWR DCD. The applicant's source term table changes are to correct errors in the originally certified ABWR DCD. Therefore, the changes are "modifications," as this term is defined in Chapter 1 of this supplement and will be evaluated using the regulations applicable and in effect at initial certification.

**Incorporation of the Condensate Storage Tank as a Radiation Source**

The following regulatory requirements provide the basis for the acceptance criteria for the staff's review:

- 10 CFR § 50.34(b)(3) (1997) required final safety analysis reports (FSARs) to include "[t]he kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposure within the limits set forth in Part 20 of this chapter."
- 10 CFR § 20.1101(b) (1997,) required that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) 61, "Fuel Storage and Handling and Radioactivity Control" (1997), required, in part, that fuel storage and handling, radioactive waste, and other systems that might contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions, including being designed with a capability to permit appropriate periodic inspection and testing of components important to safety, with suitable shielding for radiation protection, and with appropriate containment, confinement, and filtering systems.

**Clarification of Inspections, Tests, Analyses, and Acceptance Criteria**

The following regulatory requirement provides the basis for the acceptance criteria for the staff's review:

- 10 CFR § 52.47(a)(1)(vi) (1997), required DC applications to include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the tests, inspections, and analyses are performed and the acceptance criteria met, a plant that references the design is built and will operate in accordance with the DC.

Correction of Source Term Tables

The following regulatory requirements provide the basis for the acceptance criteria for the staff's review:

- 10 CFR § 50.34(b)(3) (1997) required FSARs to include "[t]he kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposure within the limits set forth in Part 20 of this chapter."
- 10 CFR § 20.1101(b) (1997) required that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.
- 10 CFR § 20.1201, "Occupational Dose Limits for Adults" (1997), required, in part, that licensees control the occupational dose to individual adults to a total effective dose equivalent of 5 rems.
- 10 CFR § 20.1601, "Control of Access to High Radiation Areas" (1997), and 10 CFR § 20.1602, "Control of Access to Very High Radiation Areas" (1997), required, in part, that licensees adequately control access to high and very high radiation areas.
- 10 CFR § 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants" (1997) and GDC 4, "Environmental and Dynamic Effects Design Bases" (1997), required that certain components important to safety be designed to withstand environmental conditions, including the effects of radiation, associated with design basis events, including normal operation, anticipated operational occurrences, and design basis accidents.
- GDC 61, "Fuel Storage and Handling and Radioactivity Control" (1997) required, in part, that fuel storage and handling, radioactive waste, and other systems that might contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions, including being designed with a capability to permit appropriate periodic inspection and testing of components important to safety, with suitable shielding for radiation protection, and with appropriate containment, confinement, and filtering systems.

### *12.4.2 Summary of Technical Information*

**Incorporation of the Condensate Storage Tank as a Radiation Source**

As originally certified, the DCD Tier 2, Section 9.2.9.2 indicated that water could be sent to the CST from several sources that could potentially be contaminated, including the control rod drive system and the radwaste disposal system. However, the ABWR DCD did not contain any source term information for the CST, nor did it describe any controls to limit effluent releases or radiation exposure from the CST during normal operations or anticipated operational occurrences, as required by 10 CFR § 50.34(b)(3) and 10 CFR § 20.1101(b). In addition, the ABWR DCD did not provide any information regarding radiation shielding for the CST or on any necessary confinement to reduce radiation exposure or to control potential leakage, in accordance with GDC 61. Therefore, to ensure compliance with 10 CFR § 50.34(b)(3), 10 CFR § 20.1101(b), and GDC 61 the staff issued an RAI letter dated September 25, 2014 (RAI) 12.02-2 (ADAMS Accession No. ML14267A352), requesting the applicant to: (1) update DCD Tier 2, Chapter 12 to provide source term and shielding information for the CST; (2) update DCD Tier 2, Chapters 11 and 12 to describe any procedures or engineering controls used to control radioactive effluents and radiation exposure from the CST, such as provisions to prevent CST overflow or design features to contain radioactive material if a leak or overflow were to occur; (3) update DCD Tier 2, Chapters 11 and 12 as appropriate, to describe the locations, functions, and design features of piping routed to and from the CST in order to ensure that radioactive effluents and radiation exposure is being adequately controlled (including design features to prevent or detect leakage from outdoor piping associated with the CST); and (4) update the radiation zone drawings in DCD Tier 2, Chapter 12, to include the location and radiation zoning for the CST.

The applicant responded to the staff's RAI and provided changes to the ABWR DCD as described below in the staff evaluation (Section 12.2.3) of this FSER supplement.

**Clarification of Inspections, Tests, Analyses and Acceptance Criteria**

The staff issued RAI 12.02-3, dated September 25, 2014 (ADAMS Accession No. ML14267A352), requesting the applicant to clarify that DCD Tier 1, Tables 3.2a and 3.2b contain DAC, instead of normal ITAAC. The applicant responded to the staff's RAI and provided changes to the ABWR DCD as requested by the staff and described below in the staff evaluation (Section 12.2.3) of this FSER supplement.

**Correction of source term tables**

DCD Tier 2, Table 12.2-3b "Gamma Ray Source Energy Spectra – Post Operation Gamma Sources in the Core" and Table 12.2-3c "Gamma Ray Source Energy Spectra – Gamma Ray Sources External to the Core During Operation," both appeared to contain errors. The source terms in both of these tables were approximately one million times lower than comparable source term tables in currently operating BWRs and in the comparable DCD for the Economic Simplified Boiling-Water Reactor (ESBWR). The NRC staff also noted that the text in DCD Tier 2, Sections 12.2.1.2.1.1.4 and 12.2.1.2.8 associated with the aforementioned tables contained different units than the units provided in the tables. For example, Table 12.2-3b used units of picojoule / Watts per second (pJ/W.s), while Section 12.2.1.2.1.1.4 indicated that the gamma ray energy spectrum was provided in joule per second per Watt (J/s/W) (neither of which appeared to be correct).

If the incorrect source term information in DCD Tier 2, Tables 12.2-3b and 12.2-3c, were to be used during plant design, significant design errors in the plant shielding design could result (suitable shielding is required under GDC 61). This could lead to an underestimation of area dose rates and higher worker doses. In this case, compliance with 10 CFR Part 20 would not be ensured because the potential design errors could result in a facility that would not be designed in accordance with the principles of maintaining occupational radiation doses ALARA (10 CFR § 20.1101) and could also potentially result in workers receiving doses in excess of the worker dose limits (10 CFR § 20.1201). Furthermore, if the incorrect source term information provided in the tables were utilized, potential high and very high radiation areas may not be properly identified (in accordance with 10 CFR § 20.1601 and 10 CFR § 20.1602). Finally, if the incorrect source term information were used in the equipment qualification analysis, the incorrect radiation exposure rates could be calculated for equipment; which would not be in accordance with 10 CFR § 50.49 and GDC 4.

Therefore, staff issued RAI 12.02-1, dated September 25, 2014 (ADAMS Accession No. ML14267A352), requesting that the applicant: (1) correct the source term data in DCD Tier 2, Tables 12.2-3b and 12.2-3c and provide documentation demonstrating the accuracy of the revised source terms; (2) update the text in DCD Tier 2, Chapter 12 to ensure the units associated with these tables were accurate; (3) ensure that the errors in the tables did not result in any other errors or inaccuracies in any other areas of the DCD, including but not limited to facility design, shielding design, radiation zoning, dose assessment, and equipment qualification; and (4) correct any additional errors identified. The applicant responded to the staff's RAI and provided changes to the ABWR DCD as described below in the staff evaluation (Section 12.2.3) of this FSER supplement.

### *12.4.3 Technical Evaluation*

**Incorporation of the Condensate Storage Tank as a Radiation Source**

In the response to RAI 12.02-2, dated January 22, 2015 (ADAMS Accession No. ML15023A016), the applicant included a combined license (COL) Information Item in DCD Tier 2, Section 12.2.3.2 to specify that the COL applicant shall determine the CST source term information (including source geometry) and provide adequate shielding to ensure the dose rate in the area surrounding the CST is less than 6 microsieverts per hour (μSv/hr), thus maintaining a radiation Zone A that allows for uncontrolled and unlimited access to the areas surrounding the CST. The applicant also proposed updating DCD Tier 2, Figures 1.2-25 and 12.3-70 to show the location of the CST and DCD Tier 2, Figure 12.3-50 to specify that the outside area adjacent to CST is designated as radiation Zone A (less than 6 μSv/hr). The staff finds it acceptable for the COL applicant to provide the source term and shielding information for the CST, based on the specific site, to allow flexibility in the liquid waste management system design. The staff also finds that it is appropriate for the CST to be shielded to maintain Zone A and to not have any access controls as described in the design for this area because of the very low dose rates of radiation Zone A. The staff notes that if any type of access controls were needed due to specific operating conditions under actual plant operation, they would be expected to be provided as part of the radiation protection program (the radiation protection program is to be provided by the COL applicant, and evaluated by staff, as provided by the COL Information Item in DCD Tier 2, Section 12.5.3.1, “Radiation Protection Program”).

In the response, the applicant also proposed updating DCD Tier 2, Section 11.2.1.2, ”Design Criteria,” which already stated that the CST has liquid level monitoring and is provided with a dike around the tank, which is routed to the radwaste system. The proposed update states that the buried portion of the CST piping will be enclosed within a pipe chase or a guard pipe and monitored for leakage. The staff finds that these means are acceptable to prevent and monitor leakage to provide assurance that any radiation exposures and unintended leakage to the environment will be kept to minimal levels.

In Supplement 1 of the response to RAI 12.02-2, dated July 7, 2015 (ADAMS Accession No. ML15194A053), the applicant proposed updating DCD Tier 2, Section 11.2.1.2 to specify that the CST dike is designed to preclude rainwater from entering the CST dike area and causing the introduction of impurities into the liquid radwaste management system, to the extent possible. The supplemental response also proposed updating DCD Tier 2, Section 11.2.1.2 to state that the structure for the transfer pumps will be integrated in the dike or the turbine building (TB), as well as the interfaces with any pipe chases. It is acceptable to locate the transfer pumps in the dike or the TB because any leakage accumulated in these areas will be collected and monitored. Also, in the supplemental response, the applicant proposed updating DCD Tier 2, Section 11.2.1.2 to specify that if leakage is detected in the pipe chase for CST piping, it will alarm in the main control room allowing operators to adequately control contamination and the release of radioactive material. The staff finds that these engineering controls are effective measures for preventing and mitigating leakage of radioactive liquid; as such, they are consistent with 10 CFR § 20.1101(b) and GDC 61 and are acceptable.

The staff finds that the proposed ABWR DCD changes described in the RAI responses and supplemental responses to RAI 12.02-2 satisfy the requirements of 10 CFR § 50.34(b)(3), 10 CFR § 20.1101(b), and GDC 61. The staff verified that the DCD changes were incorporated into the ABWR DCD Revision 6. Therefore, all expected changes regarding the responses to RAIs 12.02-2 have been incorporated into the ABWR DCD, Revision 7, and these items are closed.

Finally, in its response dated March 16, 2016 (ADAMS Accession No. ML16076A067), the applicant provided information to demonstrate that the ABWR renewal DCD meets the requirements of 10 CFR § 20.1406, “Minimization of Contamination.” The applicant proposed updating the ABWR DCD to include design features to ensure compliance with 10 CFR § 20.1406, including updating DCD Tier 2, Section 12.3.1.5.1, “Design Considerations,” to specify areas in which epoxy-type coatings are applied, which include tunnels containing piping transporting potentially radioactive contaminated liquids (including piping associated with the CST). As described in Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” epoxy coatings help to prevent leaked fluid from penetrating the tunnels and leaking into the soil. This design feature is voluntarily provided in accordance with 10 CFR § 20.1406, which did not exist at the time of initial certification. The proposed revisions associated with 10 CFR § 20.1406 have been incorporated into the ABWR DCD, Revision 7. The staff’s evaluation of compliance with 10 CFR § 20.1406 is in supplemental FSER Section 12.3 of this safety evaluation report.

**Clarification of Inspection, Tests, Analyses, and Acceptance Criteria**

In RAI 12.02-3, dated September 25, 2014, the staff asked the applicant to clarify if DCD Tier 1, Table 3.2a, Item 4 (related to compliance with 40 CFR Part 190, "Environmental Protection Standards for Nuclear Power Operations"), was appropriately classified as an ITAAC, instead of a DAC, because DCD Tier 2, Section 12.2.2.4, "Average Annual Doses," states, "For complete evaluations for compliance to 40 CFR Part 190, gamma shine evaluations are not contained in this document, since adequate detail for skyshine evaluations from the turbine complex are required in DCD Tier 1, DAC Table 3.2." In addition, all of the items in DCD Tier 1, Tables 3.2a and 3.2b are written in the form of DAC and the original ABWR FSER, Section 12.2 refers to DCD Tier 1, Tables 3.2a and 3.2b as DAC tables. Other DAC in the ABWR DCD were clearly identified in DCD Tier 1 as being DAC; however, ABWR DCD Tier 1, Tables 3.2a and 3.2b were not. The staff informed GEH that the failure to clearly identify the ITAAC as DAC was considered an error in the initial certification and should be corrected in accordance with 10 CFR § 52.57(a).

In Supplement 1 of the response to RAI 12.02-3, dated July 7, 2015, the applicant updated ABWR DCD Tier 1, Tables 3.2a and 3.2b to specify that all of the items in the tables are DAC. This is the appropriate classification for these tables, consistent with the information in DCD Tier 2, Section 12.2.2.4. Therefore, the staff finds this change to be acceptable. The staff verified that the proposed DCD changes were incorporated into ABWR DCD, Revision 6 and are reflected in the ABWR DCD, Revision 7.

**Correction of source term tables**

In the response to RAI 12.02-1, dated December 16, 2014 (ADAMS Accession No. ML14350A843), the applicant indicated that DCD Tier 2, Tables 12.2-3b and 12.2-3c contained a unit conversion error, and that the text supporting the tables in DCD Tier 2, Sections 12.2.1.2.1.1.4, "Gamma Ray Source Energy Spectra," and 12.2.1.2.8, "Radioactive Sources in the Spent Fuel," erroneously contained different units than the tables.

The applicant reviewed the data originally supporting the DCD Tier 2, Table 12.2-3b, which was initially provided in Megaelectronvolt per Watt second (MeV/W-sec) and discovered the unit conversion error in converting to the units of pJ/W-sec. The RAI response provided the initial values and converted values to show that the conversion to pJ/W-sec resulted in a conversion error of a million pJ/W-sec. The response indicated that the same error occurred in DCD Tier 2 Table 12.2-3c. Therefore, the applicant corrected the values in the tables and the supporting DCD sections to ensure that all values and units were correct. The staff verified the unit corrections and verified that the core source term values in DCD Tier 2, Tables 12.2-3b and 12.2-3c were consistent with what would be expected for a large BWR and consistent with other BWR designs. Therefore, the staff determined that the revised tables are acceptable.

In addition, GEH indicated that they reviewed the ABWR DCD to ensure that the errors did not result in any other errors or inaccuracies in the ABWR DCD.

The applicant reviewed the drywell shielding analysis supporting the ABWR DCD and the upper drywell shielding radiation dose rates with a fuel bundle on refueling bellows, shown in DCD Tier 2, Figure 12.3-74, and verified that the values in this figure were calculated using the correct source term values. The dose rates provided in DCD Tier 2 Figure 12.3-74 are comparable to dose rates for the ESBWR design, which support the applicant's conclusion that the correct source terms were used. In addition, the source term and geometry for the spent fuel pool is to be determined by the COL applicant as specified in DCD Tier 2, Tables 12.2-5a and 12.2-5b; therefore, the unit errors in the ABWR DCD source term tables did not impact the spent fuel pool design.

The applicant reviewed the worker dose estimates in DCD Tier 2, Section 12.4.1, "Drywell Dose," which provided the dose estimates for workers in the drywell. The applicant stated that these dose estimates are based on estimates of occupancy and dose rates in the drywell for typical BWRs and are not based on analytical results; therefore, the table errors had no impact on this information. The staff also reviewed the dose estimates and found them comparable to similar BWRs. In addition, radiation zoning inside containment was already labeled with the highest radiation zone designation in the ABWR DCD. Therefore, the staff determined that the table errors could not have resulted in underestimating the radiation zoning inside containment.

The applicant also indicated that there was no impact on equipment qualification. The applicant reviewed the equipment qualification dose rate values provided in DCD Tier 2, Tables 3I-7 through 3I-11, and indicated that the unit conversion errors did not impact those tables. The staff reviewed the equipment qualification dose rates in the drywell area in DCD Tier 2, Table 3I-7 and found them to be consistent with the use of the corrected source terms. Specifically, staff performed confirmatory calculations using the MicroShield computer program to estimate the gamma dose rate through the reactor shield wall and estimated a maximum dose rate in the drywell area of approximately 12 rem/hour from the reactor core. This is less than the 20 rem per hour provided by the applicant in DCD Tier 2, Table 3I-7. In addition, the staff reviewed the ESBWR DCD and found that the equipment qualification doses inside containment in the ABWR DCD are very similar to doses in the ESBWR DCD. As a result, the staff concluded that the correct source term information was used in the equipment qualification calculations.

The applicant and staff did not find any other information in the ABWR DCD that was impacted by the unit errors in the ABWR DCD tables or any other errors related to source term information in the ABWR DCD. As a result, the staff finds that the incorrect source term information in DCD Tier 2, Tables 12.2-3b and 12.2-3c did not have any impact on the plant design, equipment qualification analysis, plant radiation zoning, or worker dose estimates.

The staff verified that the applicant incorporated the proposed ABWR DCD changes described above into the ABWR DCD Revision 6 and these changes are reflected in the ABWR DCD Revision 7. Therefore, this issue is resolved.

### *12.4.4 Conclusion*

**Incorporation of the Condensate Storage Tank as a Radiation Source**

Based on the above, the staff finds that the CST design features provided in the ABWR DCD, Revision 7, meet the requirements of 10 CFR § 50.34(b)(3), 10 CFR § 20.1101(b), and GDC 61. The COL applicant will provide the CST source term and shielding information, as specified by the COL Information Item in DCD Tier 2, Section 12.2.3.2, which is acceptable.

**Clarification of Inspections, Tests, Analyses, and Acceptance Criteria**

The correction specifying that DCD Tier 1, Tables 3.2a and 3.2b, contain DAC is consistent with 10 CFR § 52.57(a) and 10 CFR § 52.47(a)(1)(vi). Therefore, the responses to RAIs 12.02-2 and 12.02-3 and associated DCD revisions are acceptable.

**Correction of Source Term tables**

Based on the above, the response to RAI 12.02-1 is complete and meets the requirements of 10 CFR § 50.34(b)(3). In addition, the table errors and subsequent correction of the errors does not invalidate any other information in the DCD and does not impact any of the staff's findings for the original ABWR certification (NUREG–1503, including Supplement 1), including those related to 10 CFR § 20.1101(b), 10 CFR § 20.1201, 10 CFR § 20.1601, 10 CFR § 20.1602, 10 CFR § 50.49, GDC 4, and 61. Therefore, the response to RAI 12.02-1 and associated DCD revisions are acceptable.

## 12.5 Radiation Protection Design Features

This evaluation documents the staff's review of the applicant's voluntary submittal to demonstrate that the ABWR design meets the requirements of 10 CFR § 20.1406(b). Since the requirements of 10 CFR § 20.1406, "Minimization of Contamination," were not applicable at the time the initial ABWR was certified, 10 CFR § 20.1406 is not required to be addressed for the renewal. However, with the supplemental information provided, the applicant chose to voluntarily comply with 10 CFR § 20.1406(b). In addition, combined license (COL) applicants referencing the ABWR design are required to conform with the operational aspects of 10 CFR § 20.1406(a) and any site-specific design information is required to address the requirements of 10 CFR § 20.1406(a).

The staff notes that the originally certified ABWR design included much of the information that would be necessary to conform to the requirements of 10 CFR § 20.1406(b). However, the applicant's supplemental information and proposed ABWR DCD revisions consolidated the information and included new design information consistent with 10 CFR § 20.1406(b) and Regulatory Guide (RG) 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning."

### *12.5.1 Regulatory Criteria*

Because the applicant's proposed design changes are voluntary, they are "amendments," as this term is defined in Chapter 1 of this supplement. Therefore, the proposed changes are evaluated using the regulations in effect at renewal. The following regulatory requirement provides the basis for the acceptance criteria for the staff's review:

- 10 CFR § 20.1406(b) requires that applicants for standard DCs submitted after August 20, 1997, describe in the application how facility design will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

### 12.5.2 Summary of Technical Information

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. In Item No. 5 of the July 20, 2012, letter, the staff requested that GEH include 10 CFR § 20.1406 design features to minimize contamination and the generation of reactor waste. In a GEH letter dated August 4, 2015 (ADAMS Accession No. ML15216A311), the applicant provided information describing how the ABWR design minimizes, to the extent practicable, contamination of the facility and the environment; facilitates eventual decommissioning; and minimizes, to the extent practicable, the generation of radioactive waste by following the guidance of RG 4.21; thereby addressing the requirements of 10 CFR § 20.1406(b). In a teleconference on January 19, 2016 (ADAMS Accession No. ML16027A283), the staff asked the applicant to address several issues regarding ABWR compliance with 10 CFR § 20.1406(b,) that were not fully addressed in the initial submittal, including adding information to the ABWR DCD on the use of embedded piping in the ABWR design and on the use of epoxy coatings. Epoxy coatings minimize the potential spread of contamination and allow for easier cleanup of spills. Embedded piping can increase the potential for undetected leaks, which could be released to the environment or result in unnecessary contamination issues when the plant is eventually decommissioned. In addition, leaks in embedded pipes can be difficult to access and repair.

Therefore, in a letter dated March 16, 2016 (ADAMS Accession No. ML16076A066), the applicant included additional supplemental information on how the ABWR is designed in accordance with 10 CFR § 20.1406(b) to address the staff's comments.

### 12.5.3 Technical Evaluation

The following evaluation addresses information provided in the August 4, 2015, GEH proposed design changes, as supplemented and clarified by the March 16, 2016, GEH submittal and the ABWR DCD proposed revisions. The evaluation also discuss some of the information already provided in the DCD, which the NRC staff determined to be acceptable information to demonstrate that the ABWR has been designed in accordance with the requirements of 10 CFR § 20.1406(b).

As part of its submittals, the applicant proposed adding ABWR DCD Tier 2, Table 12.3-8 which identifies the ABWR DCD chapter and sections that discuss implementation of the design objectives. The applicant also proposed creating DCD Tier 2, Section 12.3.1.5, "Minimization of Contamination and Radioactive Waste Generation," to provide information on how the ABWR minimizes contamination and radioactive waste generation and facilitates decommissioning, including a general description of the design and operational objectives and specific information, which are consistent with the guidance of RG 4.21. The applicant provided the following objectives:

- Objective 1 - Minimize leaks and spills and provide containment in areas where such events may occur.
- Objective 2 - Provide adequate leak detection capability to provide prompt detection of leakage from any structure, system, or component that has the potential for leakage.
- Objective 3 - Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult

(inaccessible) to conduct regular inspections (such as spent fuel pools, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination.

- Objective 4 - Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.
- Objective 5 - Facilitate decommissioning by: (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment or components that may require removal or replacement during facility operation or decommissioning.
- Objective 6 - Minimize the generation and volume of radioactive waste during operation and decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).

The GEH proposed design changes that show compliance with 10 CFR § 20.1406, include many design features consistent with the requirements of 10 CFR § 20.1406(b) and the above-mentioned design objectives. The following paragraphs discuss the significant ABWR design features for satisfying 10 CFR § 20.1406(b).

Areas where the potential for spills exists contain appropriately sloped floor drains to limit the extent of contamination. To facilitate the cleanup of leaks and spills, and to help prevent the spread of contamination, de-contaminable epoxy-type coatings are applied to both steel surfaces and concrete areas appropriate for contamination control. These areas consist of the walls and floors of the reactor building (RB) and turbine building (TB), radwaste areas, rooms containing equipment with liquid radioactive sources, floor drain areas, washdown bays, and tunnels containing piping transporting potentially radioactive contaminated liquids. In addition, equipment and floor drain sumps are lined in stainless steel to reduce crud buildup and to provide surfaces that can be easily decontaminated.

Operating experience has shown that effluent discharge piping and other underground piping can be a source of low-level environmental contamination. In particular, operating experience has shown that the following structure, system, and components (SSCs) have experienced underground piping-related events that have resulted in unmonitored, uncontrolled releases of radioactivity to the environment (i.e., condensate storage tank and associated piping, radwaste/effluent discharge piping, and cooling tower blowdown line). To the extent practical, underground piping is avoided in the ABWR design. However, the condensate storage tank (CST) piping, CST retention area drain, radwaste effluent discharge pipeline, and the cooling tower blowdown line are underground and/or contain underground piping segments. The proposed DCD updates indicate that these lines will be kept as short and direct as possible. In addition, the applicant stated that the underground piping associated with these SSCs will be designed to preclude inadvertent or unidentified leakage to the environment. This piping is enclosed within a guard pipe and will be accessible for visual inspections via a trench or tunnel. The applicant stated that threaded or flanged connections for this piping will be kept to a minimum, and other joints will be welded or otherwise permanently bonded (all piping containing radioactive material piping connections are welded to the extent practicable). Furthermore, fittings will be kept to a minimum and no in-line components will be incorporated into these lines.  These features will reduce the potential for unmonitored and uncontrolled releases to the environment and are consistent with RG 4.21 and 10 CFR § 20.1406(b).

DCD Tier 2, Section 12.3.1.2 specifies that plant equipment containing radioactive material is designed to minimize the buildup of radioactive material by minimizing the number of "dead legs" and low points.  In addition, butt-welded connections are used instead of socket welds, flanged, or screwed connections.  Butt-welded connections are stronger and less likely to leak than connection types.  To minimize trapping of radioactive crud, the design employs straight-through valve configurations, where practical, instead of valve configurations that exhibit flow discontinuities or internal crevices.  Equipment, such as heat exchangers, and piping have provisions for draining, flushing, and decontamination to minimize the generation of radioactive waste and facilitate the removal of radioactivity from crud traps.  Piping is designed to have a service life equivalent to the life of the plant.  This reduces the likelihood for leaks and also reduces potential worker dose to replace components.

Penetrations through outer walls of a building containing radiation sources are sealed to prevent miscellaneous leaks to the environment, and the process radiation monitoring system will monitor all expected radioactive release points and paths within the plant.  This minimizes the potential for unmonitored and untreated leakage from escaping the plant.  Additionally, the plant heating, ventilation, and air conditioning systems are designed to minimize airborne radiation exposures to plant personnel and releases to the environment.  These systems maintain airflow from areas of lower potential for contamination to areas of greater potential for contamination.

To facilitate decommissioning and repairs during plant operation, the RB, TB, and radwaste building are designed for large equipment removal, consisting of entry doors from the outside and numerous cubicles with equipment hatches inside the buildings.  Wherever possible, piping carrying radioactive fluids is separated from piping carrying nonradioactive fluids.  This reduces the potential for the spread of contamination.  Embedded piping will be minimized to the extent practicable, which facilitates the dismantlement of systems, reduces the likelihood of undetected leakage of radioactive fluid, and thereby facilitates decommissioning.  However, in some cases, piping is embedded, which provides radiation shielding.  As discussed above, buried piping will be kept to a minimum, and all buried piping will have features to reduce the potential for unmonitored and uncontrolled releases to the environment.

The ABWR design limits the use of cobalt-bearing materials on moving components that have historically been identified as major sources of reactor coolant contamination.  Stainless steel is used in those portions of the system that require high corrosion resistance to minimize the formation of corrosion activation products.  In addition, the COL Information Item in DCD Tier 2, Section 12.3.1.1.2 (summarized in DCD Tier 2, Section 12.3.7.4) specifies that the COL applicant will address material selection of systems and components exposed to reactor coolant to maintain radiation exposures as low as is reasonably achievable.  Therefore, the cobalt content in components in contact with reactor coolant will be minimized, which will reduce plant radiation levels and the potential spread of contamination.

Many additional design features to minimize contamination, facilitate decommissioning, and minimize, to the extent practicable, the generation of radioactive waste are described throughout the ABWR DCD. As discussed above, DCD Tier 2, Section 12.3, Table 12.3-8, provides a comprehensive crosswalk of applicable DCD chapters and sections which describe design features that address the above-listed RG 4.21 design objectives.

The NRC staff have reviewed the design features and objectives provided in the applicant's submittals and the information previously provided in the ABWR DCD and finds that these features are designed in accordance with 10 CFR § 20.1406(b), and are therefore acceptable.

In addition to the design objectives listed above, RG 4.21 contains the following operational and post-construction objectives associated with the requirements of 10 CFR § 20.1406(a):

- Periodically review operational practices to ensure that operating procedures reflect the installation of new or modified equipment, personnel qualification, and training are kept current, and facility personnel are following the operating procedures.
- Facilitate decommissioning by maintaining records relating to facility design and construction, facility design changes, site conditions before and after construction, onsite waste disposal and contamination, and results of radiological surveys.
- Develop a conceptual site model (based on site characterization and facility design and construction) that aids in the understanding of the interface with environmental systems and the features that will control the movement of contamination in the environment.
- Evaluate the final site configuration after construction to assist in preventing the migration of radionuclides offsite via unmonitored pathways.
- Establish and perform an onsite contamination monitoring program along the potential release pathways from the release sources to the receptor points.

As part of the proposed design changes, the applicant proposed adding another COL Information Item in ABWR DCD Tier 2, Section 12.3.7, "COL License Information," Section 12.3.7.5, "Requirement of 10 CFR § 20.1406," which states that the COL applicant will address the operational and post-construction objectives of RG 4.21 to meet the requirement of 10 CFR § 20.1406. The NRC staff reviewed this COL Information Item and determined that it is appropriate for the COL applicant to address the operational and post-constructive objectives of 10 CFR § 20.1406(a). Therefore, this COL Information Item is acceptable.

The NRC staff also verified that the proposed ABWR DCD changes described in the submittals were incorporated into Revision 6 of the ABWR DCD.

### *12.5.4 Conclusion*

Based on the above, the staff concludes that the ABWR DCD Revision 7, complies with the design requirements of 10 CFR § 20.1406(b). In addition, in accordance with the COL Information Item in the DCD Tier 2, Section 12.3.7.5, as discussed above, COL applicants referencing the ABWR design will be required to provide the operational and post-construction aspects of 10 CFR § 20.1406(a). As a result, the staff concludes that the ABWR DCD adequately addresses the requirements of 10 CFR § 20.1406.

# 13 CONDUCT OF OPERATIONS

## 13.1 Emergency Planning

### *13.3.1 Regulatory Criteria*

In ABWR DCD, Revision 7, which incorporated the DCD markups included in responses to requests for additional information (RAIs), GEH provided changes to the ABWR design to address various aspects of emergency planning (EP), in support of its renewal application for the ABWR standard design. These changes included revising the DCD to: (1) ensure that site-specific radiological protection for the technical support center (TSC) will be verified at the combined license (COL) application stage, consistent with the applicable TSC habitability guidance, and (2) provide for an assessment of staffing and communications capabilities to respond to a beyond-design-basis event, pursuant to certain NRC actions arising out of the NRC Fukushima Near-Term Task Force (NTTF) Recommendation 9.3. The technical justification for the changes is provided within the application, including responses to RAIs, discussed below.

As stated above, the applicant has provided ABWR DCD changes to address the TSC habitability. Since the changes are to assure compliance with the regulations in effect at the time of the original certification, consistent with the staff position at the time of original design certification, these changes are considered "modifications," as this term is defined in Chapter 1 of this supplemental FSER, and will correspondingly be evaluated using the regulations applicable and in effect at the initial ABWR certification. The following regulatory requirements provide the basis for the acceptance criteria for the staff's review:

- 10 CFR § 50.47(b)(8) (1997), required that adequate emergency facilities and equipment to support the emergency response were provided and maintained.
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 19, "Control Room," (1997), required that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.
- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.E.8 (1997), required a licensee onsite TSC and a near-site Emergency Operations Facility from which effective direction could be given and effective control could be exercised during an emergency.
- 10 CFR § 52.47(a)(1)(vi) (1997), required that a design certification application must contain the proposed inspections, tests, analyses, and acceptance criteria (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 references the design certification is built and will operate in conformity with the design certification.

In addition, the applicant has implemented ABWR DCD changes to address Fukushima NTTF Recommendation 9.3 concerning EP staffing and communications capabilities. These capabilities are not requirements and therefore are outside the scope of the DC. However, the applicant made EP enhancements that were based on the guidance provided in NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” Revision 0, issued May 2012 (ADAMS Accession No. ML12125A412). Therefore, these design changes related to non-required EP enhancements are considered an “amendment,” as this term is defined in Chapter 1 of this supplemental FSER and will correspondingly be evaluated using the regulations in effect at renewal. In this case, NTTF Recommendation 9.3 is not required by the regulations, but was evaluated by the staff to ensure consistency with the following regulatory requirements:

- 10 CFR § 50.47(b)(2), which requires that on-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.
- 10 CFR § 50.47(b)(6), which requires that provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.
- 10 CFR Part 50, Appendix E, Section IV.A, which requires a description of the organization for coping with radiological emergencies, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee’s emergency organization, and the means for notification of such individuals in the event of an emergency.
- 10 CFR Part 50, Appendix E, Section IV.E.9, which requires, in part, at least one onsite and one offsite communications system, where each system shall have a backup power source.

For the modification associated with TSC habitability, the staff determined compliance with these regulations by considering the guidance in NUREG‑0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP), Section 13.3, “Emergency Planning,” Revision 2, issued July 1981; SRP Section 6.4, “Control Room Habitability System,” Revision 2, Issued July 1981; NUREG‑0654/FEMA (Federal Emergency Management Agency)-REP-1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, issued November 1980; NUREG–0696, “Functional Criteria for Emergency Response Facilities,” issued February 1981; and Generic Letter (GL) 82-33, “Supplement 1 to NUREG–0737- Requirements for Emergency Response Capability (Generic Letter No. 82-33),” issued December 1982. For the amendment associated with NTTF Recommendation 9.3, the staff determined acceptability of the changes by considering the same guidance documents except that the staff used the SRP Section 13.3, Revision 3, issued March 2007, and assuring that the changes are consistent with regulatory requirements at initial certification.

### *13.3.2 Summary of Technical Information*

In DCD Tier 2, Section 13.3, the applicant stated that, while EP is not within the scope of the ABWR design, there are design features, facilities, functions, and equipment necessary to support EP. These design features in the ABWR Standard Plant scope include the TSC and operational support center (OSC), which are described in DCD Tier 2, Table 13.3-1, "ABWR Design Considerations for Emergency Planning Requirements." The TSC is located adjacent to the OSC (i.e., Lunch Room) in the Service Building, as shown in DCD Tier 2, Figure 1.2-19, "Control and Service Building, Arrangement Plan at Elevation 7,900 mm."

### *13.3.3 Technical Evaluation*

With regard to the TSC habitability, the staff reviewed the design description information for the Service Building Heating, Ventilation, and Air Conditioning (HVAC) System in DCD Tier 1, Section 2.15.5, "Heating, Ventilation and Air Conditioning Systems," Section 2.15.14, "Service Building," and Section 2.17.1, "Emergency Response Facilities." In addition, the staff reviewed DCD Tier 2, Section 9.4.8, "Service Building HVAC System," Section 9.4.10, "COL License Information," Section 13.3, "Emergency Planning," and Section 19A, "Response to CP/ML [construction permit/manufacturing license] Rule 10 CFR § 50.34(f)."

With regard to the Fukushima NTTF Recommendation 9.3, the staff reviewed DCD Tier 2, Section 13.3.1.2, "Staffing and Communications Capabilities," Table 1.8-21, "Industrial Codes and Standards Applicable to ABWR," and Table 1.9-1, "Summary of ABWR Standard Plant COL License Information."

**Technical Support Center (TSC) Habitability**

In NUREG–1503, Section 13.3, "Emergency Planning," the staff FSER for the original ABWR DC, the staff briefly addressed TSC habitability by stating, in part, that the TSC will contain the necessary facilities and equipment called for in Section 2, "Technical Support Center," of NUREG–0696. In addition, FSER Section 13.3 states that "[i]t is the staff's position that the facilities and equipment for the ABWR standard plant TSC should be compatible with the control room and meet the applicable criteria of NUREG–0696."

Section 2.6, "Habitability," of NUREG–0696 states, in part, that the TSC shall have the same radiological habitability as the control room under accident conditions, and the TSC ventilation system shall function in a manner comparable to the control room ventilation system. At the time of the original certification of the ABWR design, the control room radiological habitability dose criteria were 5 rem (0.05 sievert (Sv)) whole body, or its equivalent to any part of the body, as given in GDC 19. Therefore, as stated in NUREG–0737, Supplement 1, Section 8.2.1, "[TSC] Requirements," item f, the TSC should be provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. In Section 13.3 of the original ABWR FSER, the staff stated that the TSC will contain the necessary facilities and equipment called for in Section 2 of NUREG–0696, but it did not directly address whether the TSC met the habitability guidance in Section 2.6 of NUREG–0696.

In RAI 13.03-1, dated June 8, 2016 (ADAMS Accession No. ML16160A067), the staff requested that GEH address whether the TSC habitability for the ABWR standard design was consistent with the TSC habitability and ventilation system guidance in Section 2.6

of NUREG–0696 and Section 8.2 of NUREG–0737, Supplement 1. Specifically, the staff asked GEH to provide the following information:

a. Describe how the TSC ventilation system (to the extent not addressed in DCD Tier 2, Section 9.4.8) will function in a manner comparable to the control room ventilation system. For example, Section 2.6 of NUREG–0696 states that a TSC ventilation system that includes high-efficiency particulate air (HEPA) and charcoal filters are needed as minimum design features.

b. Describe how the TSC radiological habitability is the same as the control room under accident conditions, including the ABWR TSC radiological consequence analyses for the postulated DBAs [design basis accidents].

c. Revise the ABWR DCD, as appropriate, to be consistent with the TSC habitability criteria in NUREG–0696 and NUREG–0737 (Suppl. 1).

d. Add an additional ITAAC in DCD Tier 1, Table 2.17.1, "Emergency Response Facilities," to address TSC habitability, or explain why this is not necessary in this instance.

In the applicant's response dated June 28, 2016 (ADAMS Accession No. ML16180A256), GEH stated that the ABWR renewal requirements for the TSC, including habitability, remain the same as in the original ABWR design certification. GEH further explained that DCD clarity on this issue could be improved by adding an ITAAC in DCD Tier 1, Table 2.17.1, to ensure that the final as-built TSC habitability meets the commitment to NUREG–0696 in ABWR DCD Tier 2, Section 13.3. The following summarizes GEH's detailed response to Items (a.) through (d.) above.

*Response to Item (a.)*

The Service Building Clean [Area] HVAC[14] System services the TSC for personnel occupancy, and includes design features for radiological habitability. DCD Tier 2, Table 13.3-1 establishes the design considerations for the ABWR TSC radiological habitability through a reference to Section 2 of NUREG–0696 for the "necessary facilities and equipment" for the TSC. DCD Tier 2, Section 9.4.8, describes the design features for the non-safety-related Service Building Clean Area HVAC System that services the TSC that are comparable to the safety-related Control Room Habitability Area (CRHA) HVAC System, which is described in DCD Tier 2, Section 9.4.1, "Control Building HVAC."[15]

GEH also described various design considerations that affect TSC habitability, including HEPA and charcoal filters, toxic gas protection requirements, radiation shielding, and radiation monitors at the Service Building HVAC System supply air inlet. In addition, GEH identified the Service Building HVAC System ITAAC design commitments

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[14] As described in DCD Tier 1, Section 2.15.5, "Heating, Ventilating and Air Conditioning Systems," the Service Building HVAC System consists of two non-safety-related systems: (1) Clean Area HVAC System, and (2) Controlled Area HVAC System. The Clean Area HVAC System provides a controlled environment for personnel comfort and safety in the Clean Area [which includes the TSC] for the duration of a DBA.

[15] The CRHA HVAC System is also described in DCD Tier 2, Section 6.4, "Habitability Systems."

(including design criteria that support TSC radiological habitability) in DCD Tier 1, Table 2.15.5m, "Service Building HVAC System," which would be checked at the time that the COL applicant implements COL Information Item 9.4.10.1 [9.16][16] with the plant and site conditions.

In Response to Item (d.) below, GEH proposed a new ITAAC No. 6 in DCD Tier 1, Table 2.17.1, "Emergency Response Facilities," which will verify that the TSC habitability systems – including the TSC ventilation system (i.e., Service Building HVAC System) – will function in a manner comparable to the control room ventilation system.

*Response to Item (b.)*

GEH stated that because the detailed design of the non-safety-related Service Building and its HVAC systems is not yet complete, the TSC radiological consequence analyses for postulated DBAs are not included in the ABWR standard design. As noted above (in Response to Item a), through COL Information Item 9.4.10.1 [9.16], the COL applicant is to provide the details of the Service Building HVAC System, including a detailed piping and instrumentation diagram (P&ID), system flow rates, and an equipment list. This information, along with the site-specific conditions, will provide the needed information for the COL applicant to perform analyses of the TSC radiological consequences. To clarify that the COL applicant will perform the TSC radiological consequence analyses, GEH revised DCD Tier 2, Section 9.4.8.2, "System Description," and Section 9.4.10.1, "Service Building HVAC System" (COL Information Item 9.16). In addition, as described below in Response to Item (d.), GEH proposed a new ITAAC to verify TSC habitability.

*Response to Item (c.)*

As described above (in Response to Items a and b), the TSC habitability criteria (in DCD Tier 2, Table 13.3-1) are already established as being consistent with NUREG‑0696. Although the DCD does not refer to NUREG–0737, Supplement 1, as establishing criteria for TSC habitability, the Section 8.2.1, Item f, criterion is essentially the same as that established by SRP Section 6.4, "Control Room Habitability Systems," Revision 2, through reference in NUREG–0696, Section 2.6. Therefore, no revisions to DCD Tier 2, Section 13.3 are necessary in this regard because the DCD TSC habitability criteria are already consistent with NUREG–0696 (Section 2.6) and NUREG–0737, Supplement 1 (Section 8.2.1, Item f).

GEH also revised DCD Tier 1, Section 2.17.1 to add language that states: "The TSC radio-logical habitability is comparable to the control room habitability under accident conditions." This revision is consistent with the proposed new ITAAC, discussed below in Response to Item (d.).

*Response to Item (d.)*

GEH proposed a new ITAAC 6 (shown below) in DCD Tier 1, Table 2.17.1, which will verify that the TSC radiological habitability is comparable to the control room under accident conditions. ITAAC 6 reflects generic ITAAC acceptance criterion 8.1.3 in SRP

[16] DCD Tier 2, Table 1.9-1 lists COL Information Item 9.16 (Subject: Service Building HVAC System), and identifies DCD Tier 2, Subsection 9.4.10.1 as the location where a description of COL Information Item 9.16 is presented.

Section 14.3.10, Table 14.3.10-1, “Emergency Planning – *Generic Inspections, Tests, Analyses, and Acceptance Criteria (EP ITAAC)*,” Revision 3, issued March 2007.

**Table 2.17.1, Emergency Response Facilities**

ITAAC 6

- Design Commitment: *The TSC has comparable habitability to the control room habitability under accident conditions.*
- Inspections, Tests, Analyses: *An inspection of the as-built TSC habitability system will be performed, including a test of its capabilities*.
- Acceptance Criteria: *The TSC radiological habitability is comparable to the control room habitability under accident conditions such that doses to an individual do not exceed 5 rem whole body radiation exposure or 30 rem thyroid over the 30-day post-accident period.*

GEH identified various DCD design features and systems, against which the as-built TSC habitability system will be inspected and its capabilities tested. GEH also identified NUREG–0696 and SRP Section 6.4, Revision 2, as the bases for the radiological dose acceptance criteria, and made conforming changes to the Service Building HVAC System in DCD Tier 2, Sections 9.4.8.2 and 9.4.10.1 to add COL Information Item 9.16 for the Service Building HVAC System (listed in DCD Tier 2, Table 1.9-1 as Item No. 9.16). The changes state that the COL applicant will perform TSC radiological consequence analyses, considering plant and site conditions, to ensure that the TSC habitability design features ensure that doses to an individual do not exceed 5 rem (0.05 Sv) whole body or 30 rem (0.30 Sv) thyroid over the 30-day post-accident period.

With regard to performing the TSC radiological consequence analyses, the staff agrees with GEH, that consideration of plant and site conditions are needed to ensure that the doses to the TSC staff meet the radiological requirements identified above. The consideration of site conditions (as well as various final plant design features associated with the TSC that are selected by the COL applicant) are outside the scope of the certified design, such that the TSC radiological consequence analyses can only be performed at the COL application stage. Such an analysis may also require information on plant design features that is only available at the COL application stage. Therefore, the staff finds that GEH’s addition of ITAAC 6, as requested by the staff in RAI 13.03-1(d), is necessary to address TSC habitability.

In addition, the staff finds that the TSC habitability dose acceptance criteria of 5 rem (0.05 Sv) whole body and 30 rem (0.30 Sv) thyroid over a 30-day period proposed by GEH are consistent with the dose acceptance criteria given in SRP Section 6.4, Revision 2, for control room habitability, and therefore conform to the guidance in NUREG–0696, which states that under accident conditions, the TSC habitability is comparable to control room habitability. This also conforms to the guidance in SRP Section 13.3, Revision 2, NUREG–0654/FEMA-REP-1, Revision 1, and Supplement 1 to NUREG–0737. In addition, consistent with 10 CFR § 52.47, the staff finds that ITAAC 6 added to DCD Tier 1, Table 2.17.1, and the language added to COL Information Item 9.16 for the Service Building HVAC System, will ensure that the TSC habitability analyses will explicitly show that the necessary TSC radiological habitability dose criteria are met for the specific design details and site conditions pertaining to the COL application.

In Enclosure 2 to its response to RAI 13.03-1 (ADAMS Accession No. ML16180A260), GEH provided the proposed ABWR DCD markups of Tier 1, Sections 2.17 and Table 2.17-1, and Tier 2, Sections 9.4.8.2 and 9.4.10.1. The staff reviewed GEH's RAI response (described above), including the proposed ABWR DCD revisions, and finds them acceptable because they provide for the COL applicant to ensure that the TSC will have the required level of radiological protection during an emergency, consistent with the relevant guidance and the requirements of 10 CFR § 50.47(b)(8) and Section IV.E.8 of Appendix E to 10 CFR Part 50 that were applicable and in effect at the time of issuance of the original design certification.

The staff confirmed that the applicant provided the necessary information from RAI 13.03-1, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's response. Therefore, Confirmatory Item 13.3-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

The applicant also identified an additional COL Information Item in DCD Tier 2, Table 1.9-1, which relates to EP. Specifically, COL Information Item 9.16 (shown below) provides, in part (*the italicized text identifies the change to COL Information Item 9.16 in the certified DCD*), that the COL applicant will perform site-specific TSC radiological consequence analyses to ensure that the described equipment supporting the TSC provides adequate TSC radiological habitability.

**DCD Tier 2, Table 1.9-1**

| Item No. | Description | DCD Tier 2, Section |
|---|---|---|
| 9.16 | The COL applicant shall provide a detailed P&ID, system flow rates and an equipment list, compliance with RG 1.140, toxic gas protection requirements, and description of radiation monitors at the supply air inlet (if any), for the Service Building HVAC system, including the TSC and OSC, for NRC review. *The COL applicant will perform TSC radiological consequence analyses, considering plant and site conditions to ensure that TSC radiological habitability design features ensure that doses to an individual do not exceed 5 rem whole body or 30 rem thyroid over the 30-day post-accident period.* | 9.4.10.1 |

**Fukushima Dai-ichi nuclear power plant accident – NTTF Recommendation 9.3**

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. With regard to emergency planning, these design change considerations included the following request for information from Item No. 28 of the staff letter arising out of Fukushima NTTF Recommendation 9.3:[17]

> Include a COL item for Fukushima Recommendation 9.3 regarding emergency preparedness as outlined in the Request for Information pursuant to 10 CFR § 50.54(f) dated March 12, 2012 (ML12053A340).

In the March 12, 2012, Request for Information letters (RFI), NRC required all power reactor licensees and holders of construction permits to provide further information to support the evaluation of the NRC staff recommendations for the NTTF review of the accident at the Fukushima Dai-ichi nuclear facility. For NTTF Recommendation 9.3, Enclosure 5 of the RFI included guidance supporting the RFI, as well as the specific requested information.

The RFI addresses staffing and communications provisions for enhancing emergency preparedness. With regard to staffing, the accident at Fukushima highlighted the need to determine and implement the required staff to fill all necessary positions responding to a multi-unit event. Specifically, the RFI requested that all power reactor licensees and holders of construction permits (in active or deferred status) assess their current staffing levels and determine the appropriate staff to fill all necessary positions for responding to a multi-unit event during a beyond design basis natural event and determine if any enhancements are appropriate. The RFI requested single unit sites to provide the requested information, as it pertains to an extended loss of all alternating current (AC) power and impeded access to the site.

With regard to communications, the accident at the Fukushima Dai-ichi nuclear facility highlighted the need to ensure that the communications equipment relied upon to coordinate the event response during a prolonged station blackout can be powered. Specifically, the RFI requested that all power reactor licensees and holders of construction permits (in active or deferred status) assess their current communications systems and equipment used during an emergency event, including consideration of any enhancements that may be appropriate for the emergency plan. In addition, the RFI also requested consideration of the means necessary to power the new and existing communications equipment during a prolonged station blackout.

In its response letter dated July 7, 2015 (ADAMS Accession No. ML15188A269), GEH proposed a resolution of Item No. 28, and included the associated ABWR DCD markups to be included in Revision 6 of the DCD.[18] GEH's response consisted of adding COL

[17] See (1) SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A111, ADAMS Package No. ML12039A103); (2) NRC March 12, 2012, request for information associated with the NRC NTTF review of the accident at the Fukushima Dai-ichi nuclear facility (ADAMS Accession No. ML12053A340); and (3) NRC January 23, 2013, letter, which identified generic technical issues that need to be addressed as part of the NTTF Recommendation 9.3 communications capability assessment (ADAMS Accession No. ML13010A162).

[18] See also, GEH's April 29, 2016, letter, "Supplement Information for GEH's Response to Item # 26 – Fukushima Recommendation 4.2 Mitigation Strategies of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes" (ADAMS Accession No. ML16120A032), which summarizes GEH's response to NRC NTTF Recommendation 9.3 (consistent with GEH's full response in the July 7, 2015, letter) in Enclosure 2 (ADAMS Accession No. ML16120A044), Subsection 1D.2.8, "Enhanced Emergency Plan Staffing and Communication (9.3)."

Information Item 13.2a in Tier 2, DCD Section 13.3.1.2, “Staffing and Communications Capabilities,” which states the following:

> Perform an assessment as described in NEI 12-01 (Reference 13.3-1) to assess staff and communications capabilities needed to respond to a beyond design basis event.

GEH also made conforming Tier 2 changes in DCD Revision 6 that are reflected in DCD Revision 7, consisting of: (1) revising the emergency planning description in Section 13.3, “Emergency Planning,” including listing Nuclear Energy Institute (NEI) technical report NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” in Section 13.3.2, “References;” (2) listing NEI 12-01 in Table 1.8-21, “Industrial Codes and Standards Applicable to ABWR;” and (3) listing COL Information Item 13.2a in Table 1.9-1.

The staff reviewed the DCD revisions (identified above) and finds that they are acceptable because they include a COL information item for Fukushima NTTF Recommendation 9.3 regarding emergency preparedness, as outlined in the NRC’s March 12, 2012, Request for Information. This reflects the use of NEI 12-01, which the NRC endorsed in 2012 as an acceptable method for (COL) licensees to employ when addressing the RFI[19]. In addition, although these changes are not needed to meet the following regulatory requirements, the revisions are consistent with the requirements of 10 CFR § 50.47(b)(2) and (b)(6), and Sections IV.A and IV.E.9 of Appendix E to 10 CFR Part 50. Finally, the staff confirmed that the DCD changes were included in Revision 7 of the ABWR DCD. Therefore, the staff considers NRC’s July 20, 2012, request resolved, with regard to GEH’s inclusion of a COL information item in the ABWR DCD for the RFI arising out of Fukushima NTTF Recommendation 9.3 (i.e., Item No. 28 of the staff July 20, 2012 letter).

The applicant provided the requested information in regard to the Fukushima NTTF Recommendations 9.3, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant’s response letter dated July 7, 2015. Therefore, Confirmatory Item 13.3-1, from the staff advanced safety evaluation with no open items, for the ABWR DC renewal is resolved and closed.

### *13.3.4 Conclusion*

Based on the review of the applicant’s modification related to site-specific radiological protection for the TSC, and amendment related to an assessment of staffing and communications capabilities to respond to a beyond-design-basis event, the staff concludes the applicant has adequately addressed the emergency planning design-related features associated with TSC habitability, and voluntarily incorporated EP changes stemming from the NRC’s Fukushima Dai-ichi NTTF Recommendation 9.3.

[19] See (1) NRC May 15, 2012, letter, “U.S. Nuclear Regulatory Commission Review of NEI 12-01, ‘Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,’ Revision 0, dated May 2012” (ADAMS Accession No. ML12131A043), (2) NEI May 3, 2012, letter, “Transmittal of NEI 12-01, ‘Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,’ Revision 0, dated May 2012” (ADAMS Accession No. ML12125A411), and (3) NEI Report No. 12-01, Revision 0, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” May 2012 (ADAMS Accession No. ML12125A412).

Therefore, the staff concludes that the information is acceptable and meets the applicable requirements in 10 CFR § 50.47(b)(8) (1997); 10 CFR Part 50, Appendix A, GDC 19 (1997); 10 CFR Part 50, Appendix E, Section IV.E.8 (1997); and 10 CFR § 52.47(a)(1)(vi) (1997) for the modification related to TSC habitability. In addition, the information submitted is consistent with the requirements in 10 CFR § 50.47(b)(2); 10 CFR § 50.47(b)(6); and 10 CFR Part 50, Appendix E, Sections IV.A and IV.E.9 for the changes related to NTTF Recommendation 9.3. Therefore, the staff finds the changes to be acceptable.

## 13.5 Plant Procedures

### *13.5.1 Regulatory Criteria*

In the July 20, 2012 letter, the NRC identified 28 suggested changes for GEH's consideration that the staff considered to be regulatory improvements or changes that could meet 10 CFR § 52.59(b) criteria. In Item No. 17 of the letter, the NRC staff suggested that GEH update the emergency procedure guidelines and severe accident management guidelines (SAMGs) for the ABWR consistent with Nuclear Energy Institute (NEI) 91-04, Revision 1, "Severe Accident Issue Closure Guidelines," issued December 1994 (ADAMS Accession No. ML072850981). GEH responded in letters dated June 19, 2015 (ADAMS Accession No. ML15170A034) and January 30, 2017 (ADAMS Accession No. ML17031A056). GEH proposed to amend DCD Tier 2, Section 13.5, "Plant Procedures," to include additional information for combined license (COL) applicants referencing the ABWR DC, to address procedures, regarding the development of plant specific technical guidelines (PSTGs), emergency operating procedures (EOPs), SAMGs, and extensive damage mitigation guidelines (EDMGs). The changes relate to an issue that is outside the scope of the DC, and a COL applicant addressing the issue would be subject to the requirements as they exist at the time the COL application is filed. Therefore, the changes are "amendments," as this term is defined in Chapter 1 of this supplemental FSER and are evaluated using the regulations applicable and in effect at the time of renewal.

In 10 CFR § 52.79(a)(29)(ii), the NRC requires the COL applicant to submit plans for coping with emergencies, other than the plans required by 10 CFR § 52.79(a)(21). As discussed in the 2007 amendments to the 10 CFR Part 52 rule (Volume 72 of the *Federal Register*, page 49386 (72 FR 49386)), this requirement is meant to capture, for example, EOPs. The staff acceptance criteria associated with EOPs are in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 13.5.2.1, "Operating and Emergency Operating Procedures," Revision 2, issued March 2007.

The SAMGs are a voluntary industry initiative and no specific regulatory requirements govern their acceptability. While SAMGs are not a regulatory requirement, the U.S. nuclear industry has committed to developing and maintaining the SAMGs in accordance with NEI 91-04, "Severe Accident Issue Closure Guidelines," Revision 1, December 1994. Therefore, the staff reviewed GEH's changes for consistency with the current industry approach.

In accordance with 10 CFR § 52.80(d), COL applicants must provide a description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to explosions or fires, as required by 10 CFR § 50.54(hh)(2).

### *13.5.2 Summary of Technical Information*

In its letters dated June 19, 2015 and January 30, 2017, the applicant proposed to expand the COL Information Items in DCD Tier 2, Section 13.5.3.2, "Emergency Procedure Development," as stated to clarify ABWR Procedures as follows:

- Procedure development will integrate the EOPs, SAMGs, and EDMGs using industry Boiling Water Reactors Owners Group (BWROG) guidance as endorsed by applicable NRC regulatory guides (RGs) consistent with the Fukushima Near-Term Task Force (NTTF) Recommendations as described in SECY-11-0124, "Recommended Actions to be Taken Without Delay from the NTTF Report," dated September 9, 2011 (ADAMS Accession No. ML11245A127).
- Development of the PSTGs, EOPs, and SAMGs will use as inputs the standard ABWR emergency procedure guidelines described in the ABWR DCD and generic industry guidance given in NEI 91-04, Revision 1, which includes the industry commitment to incorporate severe accident strategies into the overall accident management program.
- EDMGs will be development as described in NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," Revision 2, December 2006 (ADAMS Accession No. ML070090060).
- The EOPs, EDMGs, and SAMGs will be integrated in a cohesive, effective and useable manner as described in NEI 14-01, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents," Revision 0, September 2014 (ADAMS Accession No. ML14269A236).

### *13.5.3 Technical Evaluation*

The NRC staff evaluated the changes to the COL Information Items as follows:

- The SAMGs are a voluntary industry initiative; Therefore, no regulatory requirements govern the acceptability of SAMGs. However, the staff finds that the updated COL Information Item which applies NEI 91-04 and NEI 14-01, provides that the SAMGs will be developed and maintained consistent with the current industry approach. Therefore, the staff finds the applicant's approach acceptable.
- GEH changed the ABWR DCD, COL Information Item to state that the PSTGs and EOPs will be developed using the standard ABWR emergency procedure guidelines. The staff finds this acceptable because the standard ABWR emergency procedure guidelines were reviewed and found acceptable as part of the initial ABWR certification.
- GEH changed the ABWR DCD, COL Information Item to state that EDMGs will be developed as described in NEI 06-12. Since NEI 06-02 is endorsed by DC/COL-ISG-016, "Compliance With 10 CFR § 50.54(hh)(2) and 10 CFR § 52.80(d)," the staff finds this update acceptable.
- GEH changed the ABWR DCD, COL Information Item to state that procedure development will integrate the EOPs, SAMGs, and EDMGs using industry BWROG

guidance as endorsed by applicable NRC RGs. The staff finds this acceptable since the COL applicant is directed to use NRC-endorsed approaches.

- GEH changed the ABWR DCD, COL Information item to state that these procedures and guidelines will be integrated in a cohesive, effective and useable manner as described in NEI 14-01. While not currently endorsed by the staff, the staff issued draft regulatory guide DG-1319, “Integrated Response Capabilities for Beyond-Design-Basis Events,” proposing to endorse NEI 14-01 with clarifications. Therefore, the staff finds that referencing NEI 14-01 in this COL Information Item is acceptable.

The use of COL Information Items for the ABWR DC renewal is appropriate because the overall responsibility for severe accident management is with the COL applicant and or license holder.

The applicant provided the necessary information requested by the staff in the letter dated June 20, 2012, in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant’s responses dated June 19, 2015 and January 30, 2017. Therefore, Confirmatory Item 13.5-1 from the staff’s advanced safety evaluation report with no open items for the ABWR DC renewal is resolved and closed.

### *13.5.4 Conclusion*

Based on the evaluation provided in this supplemental FSER section, the staff concludes that the design change to the ABWR DCD associated with the revision of the COL Information Items is acceptable, because it provides that the SAMGs will be developed and maintained consistent with the current industry approach and that the EOPs and EDMGs will be developed in a manner acceptable for meeting 10 CFR § 50.54(hh)(2), 10 CFR § 52.79(a)(29)(ii), and 10 CFR § 52.80(d).

# 14 INITIAL TEST PROGRAM

*14.3.2.3.6 Structural Task Group Review*

*14.3.2.3.6.1 Regulatory Criteria*

In the ABWR DCD, Revision 7, the applicant added a definition of “ASME Code” to DCD Tier 1, Section 1.1, “Definitions,” and a corresponding addition to DCD Tier 1, Section 2.1.1, “Reactor Pressure Vessel System.” This definition is consistent with the NRC position on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code at the time of the original DC, specifically the ASME BPV Code may be used contingent on the conditions imposed by the NRC in 10 CFR § 50.55a, “Codes and Standards,” including any NRC-authorized ASME Code alternatives. The addition to DCD Tier 1, Section 2.1.1 clarifies that the listed ASME materials are in Section II of the ASME Code, which is consistent with the NRC position at the time of the original design certification. As the changes are consistent with the staff position at the time of original design certification, these changes are considered a “modification,” as this term is defined in Chapter 1 of this supplemental FSER and will be evaluated using the regulations applicable and in effect at initial certification.

The applicable regulatory requirements for evaluating the ABWR DCD modification are as follows:

- 10 CFR § 52.47(a)(1)(vi) (1997) requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) necessary and sufficient to provide reasonable assurance that, if the tests, inspections and analyses are performed and the acceptance criteria met, a plant that references the design is built and will operate in accordance with the DC.
- 10 CFR § 50.55a (1997), requires compliance with codes and standards incorporated by reference into the regulations, subject to conditions imposed by the NRC and with allowance for NRC-authorized alternatives to the codes and standards.

*14.3.2.3.6.2 Summary of Technical Information*

In DCD Tier 1, Section 1.1, as supplemented by the responses to requests for additional information (RAI) described below, the applicant provided a definition of “ASME Code” to clarify that “ASME Code” refers to Section III of the ASME BPV Code unless specifically stated otherwise and that a Tier 1 departure and associated exemption is not required where Tier 1 requires compliance with the “ASME Code” and the applicant/licensee has received NRC authorization for an alternative under 10 CFR § 50.55a to Section III of the ASME BPV Code. The supplemental RAI responses discussed below add the words “Code Section II” between “ASME” and “materials” to DCD Tier 1, Section 2.1.1 to denote the specific section of the ASME Code being invoked.

*14.3.2.3.6.3 Technical Evaluation*

In the ABWR DCD, Revision 5, the applicant did not include a definition for “ASME Code,” in DCD Tier 1, Section 1.1. Without an explicit definition of “ASME Code,” a concern was raised regarding whether a combined license (COL) holder referencing a

DCD might need a Tier 1 departure and associated exemption to use an alternative to the ASME Code under 10 CFR § 50.55a. The NRC has previously stated explicitly that an exemption would not be needed for NRC-authorized alternatives to the ASME Code (as noted in the Statements of Consideration for the August 28, 2007 revision to 10 CFR Part 52, in Volume 72 of the *Federal Register*, page 49438). This reflects the NRC's historical practice of allowing use of the ASME Code contingent on the conditions imposed by the NRC in 10 CFR § 50.55a, including any NRC-authorized ASME Code alternatives. In a letter dated March 10, 2015, the staff issued RAI 14.03-1 (ADAMS Accession No. ML15068A227), due to the potential misconception that NRC-authorized alternatives to the ASME Code might be viewed as unacceptable for closure of ITAAC invoking the ASME Code. In the applicant's RAI response dated April 1, 2015 (ADAMS Accession No. ML15092A175), GEH proposed a definition, which was later supplemented by a March 2, 2017 letter (ADAMS Accession No. ML17061A065) and a March 21, 2017 letter (ADAMS Accession No. ML17080A042) after public teleconferences held on February 23, 2017, and March 16, 2017, respectively. The following is the resulting definition:

**ASME Code** means Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, unless specifically stated otherwise. Some Tier 1 content in the ABWR DCD specifies that structures, systems, and components be designed and constructed in accordance with ASME Code Section III requirements. When this language is used, it indicates that the Tier 1 requirements will be met by satisfying the edition and addenda of the ASME Boiler and Pressure Vessel Code, Section III as specified in the DCD and as incorporated by reference in 10 CFR § 50.55a subject to the conditions listed in 10 CFR § 50.55a, or in accordance with alternatives authorized by the NRC pursuant to 10 CFR § 50.55a.

In conjunction with this change, GEH added a section identifier to an ASME reference in DCD Tier 1, Section 2.1.1, where the ASME Code Section referenced was Section II instead of Section III. Because these changes do not affect previous NRC safety findings in the NUREG–1503 and NUREG–1503, Supplement 1, the staff's original FSER for the ABWR DC, or change the ABWR's compliance with ASME Code requirements, the staff finds this addition of a definition for ASME Code and a corresponding section identifier in DCD Tier 1, Section 2.1.1 acceptable. The NRC staff confirmed that the changes discussed above were incorporated in the ABWR DCD, Revision 7. Therefore, Confirmatory Item 14.03-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *14.3.2.3.6.4 Conclusion*

The NRC staff reviewed the applicant's changes to DCD Tier 1, Section 1.1, "Definitions," and DCD Tier 1, Section 2.1.1, "Reactor Pressure Vessel System." Based on the staff's technical evaluation described in this supplemental FSER section, the staff found that:

- The proposed changes do not adversely affect any previous NRC safety findings.
- The changes provided additional clarity to existing documentation.

For the reasons specified above, the staff found that the changes incorporated into the ABWR DCD, Revision 7, are acceptable.

Based on this finding, the staff concludes that the requirements of 10 CFR § 52.47(b)(1), and 10 CFR § 50.55a continue to be met with the ABWR design change as described in this supplemental FSER section.

*14.3.2.3.8 Verification of As-Built Components*

In RAI 14.03.01, dated March 10, 2015, the NRC staff asked GEH whether revisions made to the ITAAC for the Economic Simplified Boiling Water Rector DC to enhance the clarity of ASME Code requirements would be considered appropriate for the content of the ABWR DCD. Specifically, the NRC clarified the requirement for ASME Code component design verification to indicate that the activities performed to satisfy the ITAAC should be performed at the as-built stage, and they should involve a design verification and an as-built reconciliation using ASME Code design reports. In the applicant's April 1, 2015, response to RAI 14.03.01 , GEH provided confirmation to the NRC staff of its understanding that ASME Code component design verification relies on testing performed post-construction, once the as-built component is in its final installed location at the plant site, with the exception of two specific instances. In these two instances, the Reactor Pressure Vessel and Containment Vessel, the ITAAC clearly identify the documents to be reviewed. This response did not result in a change to the ABWR DCD, but the response is noted here to preserve information for future use.

*14.3.2.3.8.1 Regulatory Criteria*

The applicant does not propose any change to the ABWR DCD, but the RAI 14.03-01 response clarifies the ITAAC meaning, in that the activities performed to satisfy the ITAAC are done at the as-built stage and not during the design phase of construction. The following applicable regulatory requirements were effect at initial certification:

- 10 CFR § 52.47(a)(1)(vi) (1997), requires that a DC application contain the proposed ITAAC which 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 references the design is built and will operate in accordance with the DC.
- 10 CFR § 50.55a (1997), requires compliance with codes and standards incorporated by reference into the regulations, subject to conditions imposed by the NRC and with allowance for NRC-authorized alternatives to the codes and standards.

*14.3.2.3.8.2 Summary of Technical Information*

In the applicant's April 1, 2015, response to RAI 14.03.01, GEH confirmed its understanding that ASME Code component design verification relies on testing performed post-construction, once the as-built component is in its final installed location at the plant site, with the exception of two ITAAC, which clearly identify the documents to be reviewed.

*14.3.2.3.8.3 Technical Evaluation*

The NRC staff agrees with the applicant's response, indicating that ASME Code component design verification relies on testing performed post-construction, once the as-built component is in its final installed location at the plant site, with the exception of two ITAAC, which clearly identify the documents to be reviewed. The intent of ASME

Code component design verification is not to review as-designed components, but rather to ensure that the as-built components are consistent with the design. This is consistent with the definitions in DCD Tier 1, Section 1.1, and with the guidance in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria – Design Certification,” Draft Revision 0, issued April 1996, both of which define “as-built” as “the physical properties of the structure, system, or component following the completion of its installation or construction activities at its final location at the plant site,” and provide that a test’s purpose is to “evaluate the performance or integrity of as-built structures, systems, or components, unless explicitly stated otherwise.”

*14.3.2.3.8.4 Conclusion*

As described in the staff’s evaluation above, there are no changes to the ABWR DCD, Revision 7, but the applicant confirmed the NRC staff’s understanding of how ASME Code component design verification is accomplished. Therefore, the staff safety findings made in NUREG–1503, the staff’s FSER for the original ABWR DC, are valid and unchanged for this supplemental FSER section. The applicant’s ITAAC continue to meet the requirements of 10 CFR § 52.47(a)(1)(vi), and a COL applicant/licensee will verify that 10 CFR § 50.55a will be met for the as-built plant since it must meet the design requirements.

# 16 TECHNICAL SPECIFICATIONS

## 16.1 Regulatory Criteria

In the ABWR DCD Revision 7, the applicant proposed design changes to improve the diversity and defense-in-depth of safety systems to enhance the ABWR coping capabilities during a beyond-design-basis event. The ABWR DC renewal applicant is not required to address the mitigation of beyond-design-basis events (MBDBE) rule (10 CFR § 50.155, "Mitigation of beyond-design-basis events") that was published in the *Federal Register* on August 9, 2019 (84 FR 39684) and became effective September 9, 2019. Prior to the effective date of the MBDBE rule, the applicant provided design enhancements in its ABWR DC renewal application that would allow a potential combined license (COL) applicant the means for meeting 10 CFR § 50.155 requirements.

This evaluation documents the staff's review of the applicant's design enhancements and the proposed technical specifications (TS) changes to demonstrate that the ABWR design meets the requirements of 10 CFR § 52.47, "Contents of applications; technical information," which states that proposed TS are to be prepared in accordance with the requirements of 10 CFR § 50.36, "Technical specifications," which details the specific items (such as safety limits, limiting safety system settings, limiting conditions for operation, etc.) that must be included in the TS.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of its application to renew the ABWR DC. The applicant was requested in Item No. 26 of the July 20, 2012, staff letter to address ABWR DCD design changes related to aspects of the NRC Fukushima Near Term Task Force (NTTF) Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events based on the approach described in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (Agencywide Document and Access Management System (ADAMS Accession No. ML12039A111). Subsequently, during the MBDBE rulemaking that created 10 CFR § 50.155, the Commission decided not to impose mitigation strategies requirements on DCs.[20]

In a letter dated January 23, 2017 (ADAMS Accession No. ML17025A386), GEH revised its previous responses to Item No. 26 of the July 20, 2012, staff letter, because the MBDBE proposed rule indicated that mitigation strategies requirements would not be imposed on DCs. The applicant narrowed the scope of its changes in response to Item No. 26 to remove references to NTTF Recommendation 4.2, pending final rulemaking for the MBDBE rule. As such, GEH retained related design changes that had been proposed to address NTTF Recommendation 4.2 as well as the update to the ABWR renewal TS but requested that the NRC review these changes as operational enhancements that provide additional defense in depth. GEH stated that these proposed ABWR design enhancements could provide a potential COL applicant the means for meeting the MBDBE rule requirements of 10 CFR § 50.155. The final

[20] In the MBDBE proposed rule regulatory analysis (ADAMS Accession No. ML15266A133), the Commission proposed to not make the MBDBE proposed rule applicable to existing DCs, which included the ABWR, because "[t]he issues that may be resolved in a DC and accorded issue finality may not include operational matters, such as the elements of the [MBDBE] proposed rule."

MBDBE rule requirements did not require a change to the GEH position or proposed design changes as presented in the January 23, 2017, GEH letter.

These changes do not fall within the definition of a "modification." Therefore, in accordance with 10 CFR § 52.59(c), these design changes are "amendments," as this term is defined in Chapter 1 of this supplement and will correspondingly be evaluated using the applicable regulations in effect at renewal. Although the design related changes made by GEH are not required to meet the regulations, the staff evaluated the changes to assure that the TS remain consistent with the following regulatory requirements:

- 10 CFR § 52.47(a)(11), as relevant here, requires the applicant (GEH) to provide proposed TS prepared in accordance with the requirements of 10 CFR § 50.36.
- 10 CFR § 50.36, states that TS impose limits, operating conditions, and other requirements upon reactor facility operation for the public health and safety. The TS are derived from the analyses and evaluations in the safety analysis report. In general, the TS must contain: (1) safety limits and limiting safety system settings; (2) limiting conditions for operation (3) surveillance requirements; (4) design features; and (5) administrative controls.
- 10 CFR § 50.46, "Acceptance criteria for emergency core cooling systems [ECCS] for light-water nuclear power reactors," describes acceptance criteria for ECCS for light-water nuclear power reactors.
- 10 CFR Part 50," Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 33, "Reactor Coolant Makeup," requires a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary.
- 10 CFR Part 50, Appendix A, GDC 19, "Control Room," states, in pertinent part, that equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown.

## 16.2 Summary of Technical Information

Item No. 26 from the staff letter dated July 20, 2012, requested that the applicant address the design related aspects of Fukushima NTTF Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events as outlined in Attachment 2 of Commission Order EA-12-049 (ADAMS Accession No. ML12054A735), "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," issued on March 12, 2012.

The staff discussed NRC actions involving a pending final rulemaking for the MBDBE rule during a public teleconference held December 1, 2016. As described in the version of the draft final MBDBE rule that was publicly available at that time, no requirements would be applicable to applicants for a standard DC (or a renewal, as in the case of the ABWR application). It was also expected, at that time, that the final rule would be effective before the ABWR DC renewal would be completed. The final MBDBE rule was made effective on September 9, 2019.

On that basis, in a letter dated December 6, 2016, GEH informed the NRC of its plans to submit a revised response for addressing Item No. 26 by the end of January 2017. In its January 23, 2017, letter the applicant provided the updated GEH response for Item No. 26, maintaining some enhanced design features related to mitigating strategies that may be used by a potential COL applicant to satisfy MBDBE rule requirements including the proposed updates to the ABWR TS.

The proposed TS changes include the addition of Alternating Current (AC) Independent Water Addition (ACIWA) mode to Residual Heat Removal (RHR) Loop B (currently available for RHR Loop C), affecting TS 3.5.1, “ECCS-Operating,” and TS 3.6.2.4, “RHR Containment Spray;” and additional controls and indications on the ABWR remote shutdown panel. These additional controls and indications improve the diversity and defense in depth during beyond-design-basis events. These changes to the Remote Shutdown Panel include the following:

1. addition of wide range reactor pressure vessel (RPV) water level indication (Division I & II) (cold calibration)

2. addition of N2 supply header pressure indication (Division I & II)

3. addition of condensate storage tank (CST) water level indication (Division I)

4. addition of containment (dry well) wide range pressure indication (Division I)

5. addition of wide range suppression pool water level indication (Division I & II)

## 16.3 Technical Evaluation

**Changes to TS 3.5.1, ECCS-Operating (Add ACIWA mode to RHR Loop B (currently available for RHR Loop C):**

GEH in its submittal dated January 23, 2017, states the following regarding changes to TS 3.5.1:

> Diverse alternatives to reactor core isolation cooling (RCIC) are provided by the combustion turbine generator (CTG) and the ACIWA mode of RHR. If RCIC is inoperable, water can be injected into the RPV either by powering other ECCS subsystems from the CTG or by the fire protection system (FPS) using one of the loops of the ACIWA mode of RHR (RHR C loop or RHR B loop which is added with ABWR DCD, Revision 7).

With the RCIC inoperable and one or two inoperable ECCS subsystem(s) inoperable (Conditions B and C,) one of the loops (RHR loop B or RHR loop C) of the ACIWA mode of RHR is verified to be functional, so that the FPS can be used to inject water into the RPV during a station blackout with the RPV sufficiently depressurized. Loop B(C) of ACIWA is verified to be functional by stroking one complete cycle of each of the two manual valves in the FPS connection to the RHR Loop B(C) injection line, by starting the FPS diesel-driven fire pump and verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR Loop B(C) injection valve using its handwheel.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to the ECCS systems that provide additional capabilities and diversity in the case of a beyond design-basis event. Therefore, the staff concludes the changes are consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36, and with the staff's evaluation in the previous ABWR Final Safety Evaluation Report (FSER) NUREG–1503, Chapter 16, "Technical Specifications."

**Changes to TS 3.6.2.4, RHR Containment Spray (Add ACIWA mode to RHR Loop B (currently available for RHR Loop C):**

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), or a rapid depressurization of the RPV through the safety/relief valves, steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment (without exceeding its design pressure). The primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the wetwell airspace, bypassing the suppression pool. In that case, some means must be provided to condense steam from the wetwell so that the pressure inside primary containment remains within the design limit. This function is provided by two redundant RHR containment spray subsystems (only RHR subsystems B and C operate in this mode). The ACIWA mode of RHR provides a backup drywell or wetwell spray capability.

With one RHR containment spray subsystem inoperable, the ACIWA mode of RHR loop B or loop C, using the FPS, can be used to inject water into the drywell or wetwell spray spargers. Loop B or loop C of ACIWA is verified to be functional by stroking one complete cycle of each of the two manual valves in the FPS connection to the RHR Loop B(C) injection line, by verifying that the FPS header pressure is maintained and by stroking one complete cycle of the RHR Loop B(C) injection valve.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to the RHR systems in the case of a beyond-design-basis event. Therefore, the staff concludes the changes are consistent with the requirements of 10 CFR § 50.36 and the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16.

**Changes to TS 3.3.6.2, Remote Shutdown Panel:**

RPV wide range/narrow range water level (Addition of Wide Range RPV Water Level Indication Cold Calibration) (Div. I & II) (TS Table 3.3.6.2-1, functions 12, 13, & 27) Reactor vessel water level is provided to support monitoring of core cooling, to verify operation of the make-up pumps, and is needed for satisfactory operator control of the make-up pumps. The wide range water level channels cover the range from the near top of the fuel to near the top of the steam separators. The narrow range provides indication from near the bottom of the separators to above the steam lines. RPV level is a necessary parameter for achieving and maintaining the reactor in MODE 3. There is an additional set of wide range instruments that have been calibrated for cold conditions and will be used when the normal instruments are off scale. One channel of each of the RPV Water Level conditions and ranges is provided on each of the RSS panels. Both channels are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to the remote shutdown panel in the case of a beyond-design-basis event. Therefore, the staff concludes the changes are consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36, and with the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16.

**Suppression Pool Water Level, Narrow and Wide Range (Addition of Wide Range) (Div. I & II) (TS Table 3.3.6.2-1, functions 18, & 26)**

The suppression pool water level provides information needed to assess the status of the RCPB and to assess the status of the water supply to the ECCS. The narrow range level indicators monitor the suppression pool level from the bottom of the ECCS suction lines to five feet above the normal suppression pool level. The wide range level indicators monitor the suppression pool from the centerline of the ECCS suction piping to the wetwell spargers. One channel of both functions is provided on each of the RSS panels. Both channels are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to suppression pool water level in the case of a beyond-design-basis event. Therefore, the staff concludes the changes are consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36, and with the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16.

**Condensate Storage Tank Level (Addition of CST Water Level Indication Division I, which will be in addition to the existing Division II) (TS Table 3.3.6.2-1, functions 19)**

The CST level provides information needed to assess the status of the water supply to reactor core isolation coolant (RCIC) and high-pressure core flooder (HPCF). The indication is needed in order to achieve and maintain MODE 3 when using RCIC and HPCF. Both channels are required to be OPERABLE to achieve MODE 3 from both RSS panels.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to the CST water level indication in the case of a beyond-design-basis event. Therefore, the staff concludes the changes are consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36, and with the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16.

**N2 Header Pressure (Addition of N2 Supply Header Pressure Indication) (Div. I & II) (TS Table 3.3.6.2-1, functions 24)**

This function is provided to permit monitoring the status of the N2 bottle header pressure. These monitors are required to permit the operator to manage the N2 supply to the safety/relief valves (SRVs). One channel of this function is provided on each RSS panel. Both channels of the function are required to be OPERABLE to provide redundant capacity to achieve MODE 3 from both RSS panels.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to N2 supply pressure indications in the case of a beyond-design-basis event. Therefore, the staff concludes the change is consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36, and with the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16, "Technical Specifications."

**Drywell Pressure - Wide Range (Addition of Containment Wide Range Pressure Indication) (Div. I & II) (TS Table 3.3.6.2-1, functions 25)**

This function is provided to permit monitoring of the status of the drywell pressure. This will allow the operator to determine if there is a potential of operation of the containment overpressure protection system (COPS). One channel of this function is provided on each RSS panel. Both channels of the function are required to be OPERABLE to provide redundant capacity RSS panels.

The staff reviewed these TS changes and concludes that they are acceptable because the changes reflect the design enhancements to the containment drywell pressure indication in the case of a beyond-design-basis event. Therefore, the staff concludes the changes are consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36, and with the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16.

## 16.4 Conclusion

The staff reviewed the relevant GEH TS changes related to the ABWR design enhancements that were evaluated as ABWR DCD amendments as described in the Enclosure 1, Table 1, of the GEH letter dated January 23, 2017. As described above, these additional TS controls and indications improve the ABWR diversity and defense-in-depth during beyond-design-basis events and, therefore, enhance the safety of the plant. Therefore, the staff finds acceptable the above proposed changes made to align the TS with the ABWR DC renewal design changes. Therefore, the staff concludes that the changes are consistent with the requirements of 10 CFR § 52.47(a) and 10 CFR § 50.36 for all the associated TS changes; 10 CFR § 50.46, for the ECCS; GDC 33 and GDC 19 for light-water nuclear power reactors; and the staff's evaluation in the previous ABWR FSER NUREG–1503, Chapter 16. Therefore, the staff finds the changes to be acceptable.

# 19 SEVERE ACCIDENTS

## 19.1 <u>Probabilistic Risk Assessment</u>

### *19.1.1 Regulatory Criteria*

The applicant prepared a probabilistic risk assessment (PRA) to support the original ABWR DC rule, published May 12, 1997. The originally certified ABWR DCD did not contain this PRA but did summarize the PRA and its results. The staff reviewed and evaluated the applicant's process for updating the PRA and corresponding ABWR DCD descriptions, as appropriate, to reflect design changes made in GEH's DC renewal application.

GEH submitted the ABWR DCD, Revision 5, as part of the ABWR DC renewal application in December 2010. In the July 20, 2012 letter, the NRC staff identified 28 suggested changes for GEH's consideration that the staff considered to be regulatory improvements or changes that could meet the criteria in 10 CFR § 52.59(b). In Item Nos. 14, 15, and 16 of this letter, the staff suggested that GEH consider improving the full-power and shutdown PRA to ensure that the risk-significant structures, systems, and components (SSCs) and other risk insights are comprehensively identified. The ABWR PRA predates the improvements in PRA methods and operating experience gained since the 1997 ABWR DC rule. Therefore, the staff requested that the applicant update the ABWR PRA to fully identify the risk insights that should be used to support the identification of design and operational requirements for the DC renewal.

To renew the DC for the ABWR, the staff must find, among other things, that the design "either as originally certified or as modified during the rulemaking on the renewal" complies with the regulations in effect at initial certification. When initially certified, 10 CFR § 52.47(a)(1)(v) required the DC application to contain "[a] design-specific probabilistic risk assessment." A summary of this PRA and its results were included in the original ABWR DCD, Revision 4. To be "design-specific," the PRA and the corresponding ABWR DCD descriptions must appropriately reflect the design as it exists. Therefore, the staff determined that the impact of renewal-related design changes on the ABWR DC PRA should be adequately evaluated to determine whether the PRA requires changes.

GEH determined that the PRA and the associated ABWR DCD descriptions did not need to be changed as a result of the ABWR DC renewal-related changes. Therefore, the staff's review addressed whether a modification to the design was necessary to satisfy 10 CFR § 52.47(a)(1)(v) (1997). For this review scope, the staff evaluated the need for a modification under 10 CFR § 52.59(a), using the regulations applicable and in effect at initial certification.

As required by 10 CFR § 52.47(a)(1)(v) (1997) a DC application must contain a design-specific PRA. For safety-related SSCs, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (1997), requires, in part, that design control measures shall be provided for verifying or checking the adequacy of the design, such as by the performance of design reviews.

The staff evaluated the ABWR DC renewal PRA in accordance with NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, “LWR Edition),” (SRP) Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” Revision 3, issued December 2015. The staff used this guidance because guidance on PRA did not exist when GEH completed its PRA for the initial ABWR DC. However, the staff recognizes that GEH is not held specifically to this SRP guidance because GEH must meet the regulations applicable and in effect at initial certification.

### *19.1.2 Summary of Technical Information*

In ABWR DCD, Revision 6, submitted by letter dated February 19, 2016 (ADAMS Accession No. ML16081A268), GEH revised the ABWR DC renewal application to incorporate design changes identified in the July 20, 2012 staff letter, responses to NRC staff requests for additional information and public meetings held with the staff. In support of the safety conclusions that need to be made regarding ABWR DCD, Revision 7, Chapter 19, “Response to Severe Accident Policy Statement,” the staff conducted a PRA audit to ensure that the applicant established and conformed to an acceptable process to evaluate the impacts on the PRA due to design changes associated with its DC renewal.

The staff audited GEH documents related to renewal application design changes (from Revision 4 to Revision 6) and procedures governing engineering change control and PRA model maintenance and updates. The staff’s audit report “Regulatory Audit Results Summary Report of the Probabilistic Risk Assessment of Design Changes for the Advanced Boiling-Water Reactor Design Certification Renewal,” dated January 16, 2018 (ADAMS Accession No. ML17352A576) identifies the technical documents related to the ABWR PRA reviewed by the staff.

### *19.1.3 Technical Evaluation*

In a letter dated September 25, 2015 (ADAMS Accession No. ML15271A171), GEH stated that the PRA from the original DC remains applicable to the renewal application for Level 1 and 2 full-power risk and for shutdown risk, and that DCD Tier 2, Appendix 19K, contains a comprehensive list of risk-significant SSCs. In addition, GEH stated the following in the letter:

> GEH has established a process that requires evaluation of the design changes that are included in the renewal application. The process specifies evaluation of the changes for impact on the PRA. If a design change results in a significant impact on PRA, risk evaluation will be performed at an appropriate level.

After further evaluation, in a letter to GEH dated February 2, 2018 (ADAMS Accession No. ML17097A470), the staff determined that the suggested improvements in Item Nos. 14, 15, and 16 are not necessary for compliance with the applicable regulations in effect at initial certification and, therefore, are also not necessary for reasonable assurance of adequate protection of public health and safety. For this reason, incorporation of these suggested improvements is not necessary to support the findings required by 10 CFR § 52.59(a) to renew the ABWR DC. The staff also decided that further evaluating these improvements through the 10 CFR § 52.59(b) process is not warranted.

As described in the audit report, the staff reviewed GEH's technical information to determine if the applicant adequately evaluated and dispositioned the renewal-related design changes with respect to potential impacts on the ABWR DC PRA. Specifically, the staff reviewed the process and guidance the applicant used to assess the impact of design changes on the ABWR PRA and all of the change packages documenting the application of this process. The staff asked the applicant to conduct a table-top exercise on three design changes the staff identified as having the potential to impact the ABWR PRA. As a result of this exercise, the staff agreed with GEH's conclusion that the changes would have no significant impact on the current ABWR PRA model.

The audit allowed the staff to determine the potential impact of design changes on the ABWR design-specific PRA, confirm that the process used by GEH for PRA update meets the intent of SRP Section 19.0, and verify the applicant's compliance with its procedures. Specifically, the staff evaluated the ABWR DCD changes and their impact on the originally approved ABWR design-specific PRA and finds that the design changes have negligible impact on the PRA results including the accident sequences and frequencies that could lead to the release of radioactive fission products to the environment as described in SRP Section 19.0, "Acceptance Criteria."

The audit provided an understanding of the technical basis, assumptions, and methods by which GEH evaluates, screens, and tracks for PRA inputs or design changes. Based on its audit, the staff finds that the process used by GEH to evaluate the risk impact of design changes is acceptable and meets the intent of staff guidance in SRP Section 19.0. The applicant's conclusion that none of the GEH change packages required a change to the PRA is therefore justified. Consequently, the staff also concludes that no changes to the associated descriptions of the PRA and its results in the ABWR DCD are warranted.

### *19.1.4 Conclusion*

Based on the evaluation provided in this FSER supplement and as informed by the staff's audit of the ABWR PRA, the staff concludes that GEH has adequately evaluated and dispositioned the ABWR renewal-related design changes with respect to potential impacts on the ABWR PRA. Therefore, the staff concludes that the design as modified complies with the applicable requirements in 10 CFR § 52.47(a)(1)(v) (1997) and 10 CFR Part 50, Appendix B (1997).

### *19.2.3.3.4 ABWR Containment Vent Design*

In ABWR DCD, Revision 7, GEH identifies the ABWR containment vent system as the containment overpressure protection system (COPS). The COPS is a subsystem of the non-safety-related Atmospheric Control System (ACS). COPS is relied upon to function during beyond-design-basis events (e.g., severe accidents). The design basis of the COPS is discussed in DCD Tier 2, Section 6.2.5.2.6.2," Containment Overpressure Protection System," DCD Tier 2, Section 19E.2.8.1, "Containment Overpressure Protection System," and DCD Tier 2, Section 19K.11.6, "Containment Overpressure Protection System (COPS)." In the staff evaluation documented in Section 19.2.3.3.4 of NUREG–1503, the staff FSER for the original ABWR DC, the staff approved the COPS design for the ABWR DCD, Revision 4. In a letter dated January 8, 2016 (ADAMS Accession No. ML16008A079), GEH proposed increasing the COPS pipe diameter and rupture disk in DCD Tier 2, to reflect a correction to an error in the original system flow rate calculations, and to conform with the required minimum capacity COPS flow rate in DCD Tier 1, Section 2.14.6, "Atmospheric Control System."

#### *19.2.3.3.4.1 Regulatory Criteria*

As stated GEH made changes as reflected in ABWR DCD, Revision 7, to correct an error and to address an inconsistency between DCD Tier 1 and Tier 2. The changes to the COPS in the ABWR DCD corrects an error and inconsistency in the existing DC and therefore, they are "modifications," as that term is defined in Chapter 1 of this FSER supplement and will correspondingly be evaluated using the regulations applicable and in effect at the initial ABWR certification.

The applicant included the COPS in the original ABWR design to address Commission policy goals related to severe accidents, as documented in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," dated January 12, 1990 (ADAMS Accession No. ML003707849), and the associated staff requirements memorandum (SRM), dated June 26, 1990 (ADAMS Accession No. ML003707885), rather than to meet regulations that existed at the time of initial certification. Therefore, the staff reviewed the COPS changes to ensure the ABWR DC renewal continues to meet the Commission's position for inclusion of a dedicated containment vent path in the ABWR. In the SRM, the Commission approved the use of COPS in the ABWR subject to the results of a comprehensive regulatory review to fully weigh the potential "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that the design should provide full-flow capability to maintain control over the venting process. This Commission position was used in the review of the COPS for the original ABWR design, as discussed in Section 19.2.3.3.4 of NUREG–1503, the FSER for the original ABWR DC. The staff reviewed the COPS modifications made in the ABWR DCD, Revision 7, to ensure the conclusions reached in its review of the original ABWR DC remain valid.

#### *19.2.3.3.4.2 Summary of Technical Information*

By letter dated January 8, 2016 (ADAMS Accession No. ML16008A079), GEH submitted proposed changes to the ABWR COPS. GEH stated that during the process of confirming the detailed design of the COPS pipe diameter in an ABWR under construction, the required minimum capacity COPS flow rate of 28 kilograms per second (kg/s) specified in DCD Tier 1, could not be achieved with the certified DCD Tier 2 design information since the original system flow rate calculations did not adequately account for pipe losses. To address this issue, GEH proposed changes to the ABWR DCD to maintain the DCD Tier 1 flow rate of 28 kg/s. The design changes increase the diameter of the COPS piping from 250 millimeters (mm) (10 inches (in)) to 350 mm (14 in), and the rupture disk size from 200 mm (8 in) to 250 mm (10 in). GEH states that these DCD Tier 2 design changes to correct the flow rate calculation error will achieve the minimum COPS flow rate of 28 kg/s as described in DCD Tier 1, Section 2.14.6.

In letters dated February 18, 2016 (ADAMS Accession No. ML16049A044), April 19, 2016 (ADAMS Accession No. ML16110A154), June 16, 2016 (ADAMS Accession No. ML16168A302), and October 11, 2016 (ADAMS Accession No. ML16285A132), GEH provided additional ABWR DCD changes and supporting technical information in regard to the COPS.

*19.2.3.3.4.3 Technical Evaluation*

The staff approved the COPS design as a part of the original ABWR DC review which is documented in Section 19.2.3.3.4 of NUREG–1503, the staff FSER for the original DC. The staff review of the DC renewal COPS design changes focused on assessing the potential impacts on the COPS performance analyses in support of the originally certified ABWR design and associated staff findings.

The staff reviewed the applicant's changes in Chapters 6, "Engineered Safety Features," and Chapter 19, "Response to Severe Accident Policy Statement," of the ABWR DCD, Revision 6, to determine if the COPS is able to meet the required minimum flow rate specified in DCD Tier 1, Section 2.14.6. Fluid flow in piping is accompanied by friction and this friction is reflected in the system performance (e.g., flow rate). In a system with a given pressure input (e.g., actuation pressure for a rupture disk), as the system friction decreases (e.g., due to increasing pipe diameter), the flow rate increases. Therefore, the staff determined that GEH's approach to increase the size of piping and components in the COPS flow path is a reasonable approach to reduce friction losses in order to meet the required minimum flow capacity for COPS.

A staff analysis confirmed that the applicant's original system flow rate calculations did not properly account for pipe losses. In a public teleconference on September 22, 2016 (ADAMS Accession No. ML17004A315), the staff requested that GEH provide details on the calculated piping losses. In its letter dated October 11, 2016 (ADAMS Accession No. ML16285A132), GEH provided the overall system resistance coefficient for its calculations. The staff review and analysis confirmed that the applicant's revised COPS design and analysis properly account for pipe losses and provide assurance that the COPS flow requirement in DCD Tier 1, Section 2.1.4 is met. In addition, DCD Tier 1, Table 2.14.6, "Atmospheric Control System," Inspections, Tests, Analyses and Acceptance Criteria 2.14.6-04, requires a combined license holder to confirm that the design meets the Tier 1, Section 2.1.4, COPS flow rate based on the as-built plant layout and the as-built system loss coefficients. Therefore, staff finds the applicant's DCD Tier 2, changes meet the DCD Tier 1, COPS flowrate acceptance criterion.

In a supplemental letter dated April 19, 2016 (ADAMS Accession No. ML16110A154), GEH stated that they reviewed the applicable severe accident analyses and evaluations performed for the ABWR DC and identified those items potentially affected by the COPS design changes. Based on the review, GEH identified the impact of flashing during venting to be the only analysis affected by the change, particularly related to the suppression pool surface response to a decompression wave. Since the increase in the COPS sizing maintains the original COPS performance characteristics (e.g., rupture disk setpoint of 0.72 Megapascal (MPa), minimum COPS flow rate of 28 kg/s), the staff determined that the original analyses, other than the COPS pressure loss error, remain valid, including the thermal-hydraulic accident sequence and core melt progression analyses.

In DCD Tier 2, Section 19E.2.3.5.1, "Response of Suppression Pool Surface to Decompression Wave," GEH describes the evaluation of the response of the suppression pool surface to a decompression wave. DCD Section 19E.2.3.5.1 states that the decompression resulting from the COPS rupture disc opening during an accident is not large enough to cause pool pressure to drop below its saturation pressure of 330 kilopascals (kPa) at its initial temperature of 410 kelvin (K), or 137 degrees Celsius °(C) (738 degrees Rankine (R) or 278 degrees Fahrenheit °(F)) and that the pool surface would move upward at only a negligible velocity for the transmitted decompression. GEH reevaluated the response of the suppression pool surface to a decompression wave for the new COPS piping and the rupture disk sizes to confirm that the conclusions in the certified ABWR DCD are unchanged and provided associated changes to DCD Tier2, Section 19E.2.3.5.1.

The staff review identified errors in three equations in DCD Tier 2, Section 19E.2.3.5.1.2, "The Gas Discharge Rate," Equations 19E.2-41a, 19E.2-41d, and 19E.2-41k. The (k + 1) term in the denominator of the exponential term in Equations 19E.2-41a and 19E.2-41d should be corrected as (k – 1). In these equations, k represents the specific heat ratio for an ideal gas because the applicant assumed that the nitrogen and steam mixture in the COPS piping would behave as an ideal gas. To check the validity of this assumption the staff calculated the suppression pool surface response to a decompression wave using thermodynamic properties of a mixture of nitrogen and saturated steam instead. The magnitude of the decompression wave transmitted into the water pool and the resulting pool rise velocity, as presented in DCD Tier 2, Section 19E.2.3.5.1.6, "Water Dynamic and Thermodynamic Response," did not change (up to two significant figures). Therefore, the staff determined that applicant's assumption is acceptable. The Ct/r in the exponential term in Equation 19E.2-41k should be corrected as Ct/R. In this equation, C, t, r, and R represent the acoustic speed, time, distance from the center of COPS piping at the entrance to the suppression pool, and radius of COPS piping at the entrance to the suppression pool, respectively. The staff found that these errors did not affect the results provided in the certified ABWR DCD or in the DC renewal application. In its calculations GEH used the corrected form of Equations 19E.2-41a and 19E.2-41d and a conservatively simplified version of Equation 19E.2-41k that did not contain the erroneous term.

In the applicant's letters dated February 18, 2016 (ADAMS Accession No. ML16049A043), and June 16, 2016 (ADAMS Accession No. ML16168A301), GEH stated that after its review, it agreed with the staff on these errors. Corrected the equations and provided proposed DCD changes in ABWR DCD, Revision 6, markups.

The staff's review of GEH's February 19, 2016, and June 16, 2016, letter submittals confirms that the increase in COPS piping and the rupture disk sizes did not affect the conclusions in the certified ABWR DCD that; (1) the decompression resulting from the COPS rupture disk opening during an accident is not large enough to cause pool pressure to drop below its saturation pressure of 330 kPa at its initial temperature of 410 K or 137°C (738 R or 278°F), and (2) that the pool surface would move upward at only a negligible velocity for the transmitted decompression. The staff's review also finds GEH's changes to DCD Tier 2, Section 19E.2.3.5.1 acceptable.

The applicant provided the necessary information as described above in the ABWR DCD, Revision 7, which incorporated the changes described in the applicant's letter submittals. Therefore, Confirmatory Item 19.02-1 from the staff's advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *19.2.3.3.4.4 Conclusion*

Based on the evaluation provided in this supplemental FSER section, the staff concludes that the changes do not alter the safety findings made in Section 19.2.3.3.4 of NUREG–1503, the FSER for the certified ABWR DC and remain consistent with the Commission's position for inclusion of a dedicated containment vent path in the ABWR, as documented in SECY-90-016 and the associated SRM. Therefore, the staff finds that the design, as modified, satisfies the NRC's regulations applicable and in effect at initial certification.

## 19.5 <u>19.5(A) Aircraft Impact Assessment</u>

10 CFR § 52.59(a), "Criteria for renewal," states, in part, that the first time the Commission issues a rule granting the renewal for a standard DC in effect on July 13, 2009, the Commission shall find that the renewed design complies with the applicable requirements of 10 CFR § 50.150, "Aircraft Impact Assessment," the aircraft impact assessment (AIA) rule. The ABWR DC for which GEH is requesting renewal was in effect prior to July 13, 2009. Therefore, the applicant's design changes to address these requirements are reflected in the ABWR DCD, Revision 7.

The impact of a large, commercial aircraft is a beyond-design-basis event (BDBE). Under 10 CFR § 50.150 DC renewal applicants for new nuclear power reactors are required to perform a design specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Applicants are required by 10 CFR § 50.150(b) to submit a description of the design features and functional capabilities identified as a result of the assessment in its DCD, along with a description of how the identified design features and functional capabilities show that they meet the acceptance criteria in 10 CFR § 50.150(a)(1).

The Statement of Considerations for the AIA rule regarding new nuclear power reactors states that: "The NRC's decision on an application subject to 10 CFR § 50.150 will be separate from any NRC determination that may be made with respect to the adequacy of the impact assessment which the rule does not require be submitted to the NRC." Since the AIA is not submitted to the NRC for its review, the staff conducts its DC review to determine whether or not descriptions of the design features and functional capabilities are complete enough such that, assuming the design features and functional capabilities perform their intended functions, there is reasonable assurance that the acceptance criteria in 10 CFR § 50.150(a)(1) can be met.

This ABWR DC renewal supplemental FSER section describes the staff's evaluation of the applicant's DCD Tier 2, Section 19G, "Aircraft Impact Assessment," and changes to the ABWR DCD, Revision 7.

### *19.5.1 19.5(B) Regulatory Criteria*

As described in Section 19.5(A) of this supplemental FSER Section, 10 CFR § 52.59(a) and 10 CFR § 50.150 require renewal applicants to perform a design-specific assessment of the effects on the facility resulting from the impact of a large, commercial aircraft for new nuclear power reactors. The applicant has made changes in ABWR DCD, Revision 7, with a description of the design features and functional capabilities identified as a result of the assessment in its ABWR DCD, along with a description of how the identified design features and functional capabilities show that the acceptance criteria in 10 CFR § 50.150(a)(1) are met. Therefore, in accordance with 10 CFR § 52.59(a), this design change is required by the Commission as discussed in Chapter 1 of this supplemental FSER and will correspondingly be evaluated using the regulations in effect at renewal.

*19.5.2 19.5(B).1 Applicable Regulations*

The staff used the following relevant regulations and guidance to perform this review:

- 10 CFR § 50.150(a)(1), requires that applicants perform a design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (i) the reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or spent fuel pool (SFP) integrity is maintained.
- 10 CFR § 50.150(a)(3)(iii)(B), states that the requirements of 10 CFR § 50.150(a)(1) and (a)(2) shall apply to applicants for renewal of standard DCs in effect on July 13, 2009, that have not been amended to comply with the requirements of 10 CFR § 50.150 by the time of application for renewal.
- 10 CFR § 50.150(b), requires that the final safety analysis report describe (1) the design features and functional capabilities that the applicant has identified for inclusion in the design to show that the facility can withstand the effects of a large, commercial aircraft impact in accordance with 10 CFR § 50.150(a)(1) and (2) how those design features and functional capabilities meet the assessment requirements of 10 CFR § 50.150(a)(1).

*19.5(B).2 Review Guidance*

- NRC Regulatory Guide (RG) 1.217, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts," Revision 0, issued August 2011 (ADAMS Accession No. ML092900004), provides guidance for applicants to demonstrate compliance with NRC regulations with regard to AIA. In particular, this RG endorses the methodologies described in the industry guidance document, Nuclear Energy Institute (NEI) 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 8, issued April 2011 (ADAMS Accession No. ML111440006).
- NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 19.5, "Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts," Revision 0, issued April 2013, provides guidance for meeting the requirements in 10 CFR § 50.150(a).

*19.5.2 19.5(C) Summary of Technical Information*

In DCD Tier 2, Section 19G, the applicant stated that an AIA was performed in accordance with the requirements in 10 CFR § 50.150(a)(1) using the methodology described in NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0, and SRP Section 19.5, Revision 0. Based on the results of its assessment and staff feedback concerning AIA security-related and proprietary information during a non-public teleconference held on January 19, 2017 (ADAMS Accession No. ML17013A018), the applicant identified a set of key design features to show that the acceptance criteria in 10 CFR § 50.150(a)(1) are satisfied. The applicant submitted these key design features in ABWR DCD, Revision 6, and in ABWR DCD, Revision 6, markups, based on information submitted in the applicant's letter dated February 28, 2017 (ADAMS Accession No. ML17059C517). The applicant's letter contains Technical Report (TR) NEDE-33875, Revision 3, issued February 2017 (ADAMS Accession Nos. ML17059C523 (public version) and ML17059C525 (proprietary version)), which is incorporated by reference into the DC renewal application and will be part of the renewed ABWR DC. In addition, the technical report references other sections of the ABWR DCD that provide additional details in support of the ABWR AIA. DCD Tier 2, Section 19G, also describes how the key design features show that the acceptance criteria in 10 CFR § 50.150(a)(1) are met.

#### *19.5(C).1 Description of Key Design Features*

DCD Tier 2, Section 19G, describes the credited design features, functions, and references to sections containing the detailed descriptions as summarized below:

- DCD Tier 2, Section 19G.4.1, "Primary Containment"
  (1) The reinforced concrete containment vessel (RCCV) as described in DCD Tier 2, Sections 3.8, "Seismic Category I Structures," and Section 3H.1, "Reactor Building," protects the safety systems located inside primary containment from the impact of a large, commercial aircraft.
- DCD Tier 2, Section 19G.4.2, "Site Arrangement and Plant Structural Design"
  (1) The location and design of the control building (C/B) structure as described in DCD Tier 2, Section 3.8.4, "Other Seismic Category I Structures," and DCD Tier 2, Section 3H.2 "Control Building," protect portions of the reactor building (R/B) from the impact of a large, commercial aircraft. The C/B location, fixed with respect to other major structures, is defined in GEH TR NEDE-33875P, Revision 3, to ensure that credit of the C/B as an intervening structure is maintained.

  (2) The location and design of the turbine building (T/B) structure and layout as described in DCD Tier 1, Section 2.15.11, "Turbine Building," and DCD Tier 2, Figures 1.2-24 through 1.2-31, protect the entire north wall of the C/B and portions of the north wall of the R/B from the impact of a large, commercial aircraft. The T/B location, fixed with respect to other major structures, is defined in GEH TR NEDE-33875P, Revision 3 to ensure that credit of the T/B as an intervening structure is maintained.

  (3) The location and design of the R/B structure as described in DCD Tier 2, Sections 3.8.4 and DCD Tier 2, Section 3H.1, "Reactor Building," protect portions of the primary containment and the entire south wall of the C/B from the impact of a large, commercial aircraft. This includes the protection provided by exterior walls, interior walls, intervening structures and barriers on the large openings in the reactor building exterior walls. The reactor well shield plugs protect the drywell head from secondary impacts as identified in DCD Tier 2, Section 3H.1.3, "Structural Description." The R/B location, fixed with respect to other major structures, is defined in GEH TR NEDE-33875P, Revision 3 to ensure that credit of the R/B as an intervening structure is maintained.

(4) The location and design of the SFP and its supporting structure as described in DCD Tier 2, Section 9.1, “Fuel Storage and Handling,” and DCD Tier 2, Figure 1.2-12, “Reactor Building, Arrangement Plan at Elevation 31700/38200 mm,” protect the SFP from the impact of a large, commercial aircraft.

(5) The physical separation of the Class 1E emergency diesel generators prevent the loss of all electrical power to core cooling systems by protecting them from physical damage, fire damage and smoke effects.

(6) The location and design of the service building (S/B) structure as described in DCD Tier 2, Section 3H.6, “Summary of Key Structural Design Features,” and Figures 1.2-20 through 1.2-22 protect the east wall of the C/B from the impact of a large, commercial aircraft. The S/B location, fixed with respect to other major structures, is defined in GEH TR NEDE-33875P, Revision 3 to ensure that credit of the S/B as an intervening structure is maintained.

(7) The location and design of the C/B annex structure as described in DCD Tier 2, Section 3H.6 and DCD Tier 2, Figures 1.2-20 through 1.2-22, protect the west wall of the C/B from the impact of a large, commercial aircraft. The C/B annex location, fixed with respect to other major structures, is defined in GEH TR NEDE-33875P, Revision 3, to ensure that credit of the C/B annex as an intervening structure is maintained.

(8) The seismic gap between the R/B and C/B described in DCD Tier 2, Section 3.8.5.1, “Description of the Foundations,” protects the C/B from shock effects from strikes on the R/B.

(9) The R/B heating ventilation and cooling system (HVAC) ducting locations ensure routing maintains separation divisionally through protection or physical separation so that the impact of a large, commercial aircraft strikes do not result in a loss of all divisions of core cooling.

(10) During normal operating conditions, the R/B crane will be parked at the R/B north wall when not in use.

(11) Any permanent structure that penetrates the C/B roof is sized to preclude a strike from the east and west directions.

- DCD Tier 2, Section 3H.6, “Summary of Key Design Features”

(1) Structural configuration of the SFP within the R/B precludes a direct strike on the SFP. The SFP is a reinforced concrete structure with an American Society for Testing and Materials (ASTM) A-240 Type 304L stainless steel liner. The SFP walls are strengthened as described in TR NEDE-33875P, Revision 3 to ensure that the structural integrity of the SFP is maintained.

(2) Structural configuration of the RCCV within the R/B precludes a direct strike on containment, and the structural design of the RCCV ensures that the RCCV is not perforated.

(3) Shield blocks over the drywell head are to be configured to fully resist secondary impact from concrete debris, aircraft wreckage, and falling crane components to protect the integrity of drywell head. The reactor cavity shield blocks are shown in DCD Figure 3H.1-23, “Reactor Building Reactor Cavity Shield Blocks.”

(4) Interior partition walls are to be thickened and strengthened as shown in TR NEDE-33875P, Revision 3, to limit physical damage to interior partition walls.

(5) Reinforced concrete sliding barriers with structural capacity equivalent to the surrounding wall are to be provided for the 6 large openings on 1F (DCD Tier 2, Figure 1.2-8, "Reactor Building, Arrangement Plan").

(6) Protective awnings for the HVAC exhaust openings on 2F (DCD Tier 2, Figure 1.2-9, "Reactor Building, Arrangement Plan,") are sized to provide structural capacity equivalent to the corresponding exterior wall to prevent unabated wreckage through these openings.

(7) Protective awnings for the HVAC intake openings on 3F (DCD Tier 2, Figure 1.2-10, "Reactor Building, Arrangement Plan") are sized to provide structural capacity equivalent to that provided in Table 3-2 of NEI 07-13, Revision 8 for exterior walls.

(8) Deleted.

(9) The exterior walls of the C/B annex are to be reinforced concrete.

(10) The exterior wall of the S/B is to be reinforced concrete.

(11) The exterior wall of the T/B is to be reinforced concrete.

(12) The exterior walls of the R/B on the east, west, and south sides are to be strengthened with enhanced reinforcement as described in TR NEDE-33875P, Revision 3.

*19.5(C).2 Description of How Regulatory Acceptance Criteria are Met*

The acceptance criteria in 10 CFR § 50.150(a)(1) require the applicant to perform a design-specific assessment of the impact of a large, commercial aircraft on the facility. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (i) the reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or SFP integrity is maintained.

In the ABWR DCD, Revision 6, the applicant in DCD Tier 2, Section 19G, and in ABWR DCD, Revision 6, markups as submitted by letter dated September 2, 2016 and November 23, 2016, GEH indicates that it meets the 10 CFR § 50.150(a)(1) acceptance criteria by including features in the ABWR design that, following the impact of a large, commercial aircraft, show that the design can:

- Maintain core cooling, and
- Maintain SFP integrity.

The applicant's ABWR AIA, to maintain core cooling and SFP integrity, credits the safety-related systems as described in DCD Tier 2, Section 19G of the ABWR DCD, Revision 7. The ABWR DC renewal AIA design changes ensure that the reactor can be shut down and decay heat can be adequately removed from the reactor core following the impact of a large, commercial aircraft. The key design features and physical separation for assuring core cooling are described in DCD Tier 2, Section 6.3, "Emergency Core Cooling." Some of this equipment is located inside the RCCV and some is located inside the R/B. Locations inside the RCCV are protected from structural, shock, and fire damage by the design of the RCCV structure as well as the R/B structure, which limits the penetration of a large, commercial aircraft such that the

RCCV is not perforated. Equipment inside the R/B is protected by structural design features of the R/B itself and by structures adjacent to the R/B, including the T/B, the C/B annex, and the S/B. In addition, fire barriers have been designed and located in the R/B to contain the spread of fire inside the building such that at least one train of safety-related equipment for core cooling is protected for each R/B impact scenario.

As for maintaining spent fuel integrity, GEH provided design changes in DCD Tier 2, Section 19G.5, which determined that impact from a large, commercial aircraft would not result in perforation of the SFP liner, and no SFP liner leakage, or SFP drain down conditions would occur from piping attachments that would result in leakage below the required minimum SFP water level.

### *19.5.3 19.5(D) Technical Evaluation*

The staff reviewed the AIA information in DCD Tier 2, Section 19G, and the referenced DCD sections and evaluated the following:

#### 19.5 (D).1 Reasonably Formulated Assessment

In DCD Tier 2, Section 19G, the applicant stated, that its AIA is based on the guidance provided by RG 1.217, Revision 0, and NRC endorsed NEI 07-13, Revision 8, with no exceptions. The staff also finds that the applicant’s hired contractors used to perform the AIA were well-experienced and have performed the AIA previously for other design centers.

The staff finds that the applicant adequately meets the guidance in SRP Section 19.5, Items III.1 and 2, because the applicant used an assessment methodology that conforms to the guidance in NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0, and the assessment was performed by qualified personnel consistent with the guidance in SRP Section 19.5, Item III.2.

#### 19.5(D).2 Key Design Features for Core Cooling

In DCD Tier 2, Section 19G.4.4, “Core Cooling Features,” the applicant described the key design features for assuring core cooling. The staff’s evaluation of these key design features is documented in other sections of NUREG–1503 the FSER for the original ABWR DC. For example, FSER Section 9.2.1.5 evaluates the reactor service water system and FSER Section 9.2.11 describes the reactor building cooling water system. These systems are key design features for providing the necessary cooling water for ABWR emergency core cooling system (ECCS) operation. The staff used the information provided by the applicant to confirm that these features are also suitable for maintaining core cooling following the impact of a large, commercial aircraft. During the review, the staff also confirmed that all these design features can be initiated and operated from the control room or an alternate location, and require little, if any, further operator intervention to maintain the core cooling function.

The applicant stated that, following normal power operation, an undamaged ECCS has the capability of maintaining core cooling. In addition to the ECCS, the applicant identified support systems necessary to maintain core cooling. Table 19.5 of this FSER Supplement shows the staff’s compiled list of the credited key design features identified in ABWR DCD Tier 2, Section 19G.

The applicant's assessment determined that at least one division of ECCS would be available following the impact of a large, commercial aircraft on the R/B. The applicant credited advance warning, consistent with NEI 07-13, Revision 8, for the operators to take manual action to shutdown the reactor prior to impact. The applicant further described that the hydraulic control units are located below grade, outside of the assessed AIA damage footprint of the ABWR design. The applicant further described that during shutdown conditions (reactor shutdown with the reactor head removed and reactor water level at the level of the vessel flange or higher) administrative controls will be established by the combined license applicant to ensure residual heat removal (RHR) train A and either RHR or high pressure core flooder for train B and C are not out of service for maintenance until the cavity is flooded. This will ensure an adequate water reservoir to provide cooling of the fuel in the vessel for at least 24 hours.

The staff reviewed changes to the ABWR DCD as proposed in a GEH Letter dated September 2, 2016 (ADAMS Accession No. ML16258A350), and its supplement dated November 23, 2016 (ADAMS Accession No. ML16334A292). The applicant provided ABWR DCD, Revision 6, markups, drawings, and a TR NEDE- 33875P, Revision 3, necessary to update the ABWR design to the latest AIA. These letters identify additional key design support features for core cooling. For example, the letters state that cabling and ventilation is routed divisionally, and the main control room HVAC mechanical and electrical cross connects are identified as key design features for core cooling. The staff finds the applicant's addition of key design features acceptable because it modifies the ABWR DCD to contain a description of the design features and functional capabilities as required by 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

Based on the staff's review of DCD Tier 2, Section 19G, and the applicant's use of the NRC endorsed guidance document NEI 07-13, Revision 8, the staff finds that the applicant performed a reasonably formulated AIA analysis that identifies key design features necessary for core cooling. Also, based on the above, the staff finds the applicant's description of the key design features for maintaining core cooling to be adequate and acceptable, and therefore meets the requirements of 10 CFR § 50.150(b).

Table 19.5 provides a complete list of the ABWR key design features as shown below.

**Table 19.5-1 Key Design Features**

| Design Feature | DCD Reference Sections | Function |
|---|---|---|
| Fire Barriers: 3-hour fire-rated | 9.5.1<br>9A | Protect core cooling equipment from fire damage |
| Fire Barriers: 3-hour fire-rated, 5-psid rated | 9.5.1<br>9A | Protect core cooling equipment from fire damage |
| Emergency Core Cooling Systems | 6.3 | Core cooling |
| Reactor Service Water System | 9.2.15 | Core cooling |

| Design Feature | DCD Reference Sections | Function |
|---|---|---|
| Reactor Building Cooling Water System | 9.2.11 | Core cooling |
| Class 1E ac and dc Power Systems | 8.3.1; 8.3.2 | Core cooling |
| Instrumentation System | 7.2; 7.3.2.1; 7.3.2.4; 7.3.2.6; 7.3.2.7; 7.3.2.8 | Core cooling |
| AC Independent Water Addition System | 5.4.7 | Core cooling |
| Control Rod Drive Hydraulic Control Units | 4.6.1 | Core cooling |
| Ultimate Heat Sink | 9.2.5 | Core cooling |
| Containment Overpressure Protection System | 6.2.5 | Core cooling |
| Reactor Safety Relief Valves | 6.2 | Core cooling |
| Main Control Room HVAC | 9.4.1.1.4 | Core cooling |
| Reactor Building HVAC | 9.4; Appendix 9A | Core cooling |
| Makeup Water Condensate System | 9.2.9 | Core cooling |
| Fire Water Storage System | 9.5.1 | Core cooling |
| Suppression Pool | 6.2.1 | Core cooling |
| SFP and Support Structures | 9.1 and Figure 1.2-12 | SFP Integrity |
| Primary Containment | 3.8; 3H.1 | Protect core cooling equipment |
| Control Building | 3.8.4; 3H.2 | Protect core cooling equipment and provide screening for reactor building |
| Turbine Building | Tier 1 2.15.11; Figure 1.2-24 through 1.2-31 | Provide screening for control building and reactor building |
| Control Building Annex | 3H.6; Figures 1.2-20 through 1.2-22 | Provide screening for control building |
| Service Building | 3H.6; Figures 1.2-20 through 1.2-22 | Provide screening for control building |
| Reactor Building | 3.8.4; 3H.1 | Protect core cooling equipment and SFP integrity, and provide screening for the control building |

## 19.5(D).3 Key Design Features that Protect Core Cooling Design Features

The key ABWR design features and functional capabilities that protect the core cooling design features are described below. These include: fire barriers and fire protection features, plant arrangement and plant structural design features, ability to survive shock-induced vibrations, and ability to trip the reactor.

### *19.5(D).3.1 Fire Barriers and Fire Protection Features*

In DCD Tier 2, Section 19G.4.3, “Fire Barrier and Fire Protection Features,” the applicant identified and described the fire protection key design features that protect core cooling equipment. These include the design and location of the 3-hour fire rated fire barriers and the 5 pounds-per-square-inch-differential (psid) (34.5 kilopascal (kPa)), 3-hour fire rated barriers within the R/B. The applicant indicated that the assessment credited the design and location of the R/B fire barriers (including floor assemblies, doors, penetration seals, and dampers) as described in DCD Tier 2, Sections 9.5.1 and 9A.4 (which includes Figures 9A.4-1 through 9A.4-10). These fire barriers limit the effects of internal fires created by the impact of a large commercial aircraft. The applicant clarified that all credited water-tight doors will have a 5 psid (34.5 kPa), 3-hour fire rating. Additionally, all credited penetration seals in 3-hour fire barriers will also be rated for 3-hour, 5-psid. Fire dampers with a 3-hour 5-psid rating will be quick actuating (blast) type.

In addition, the staff reviewed the fire protection related changes to the ABWR DCD, Revision 6. As a result of preparing for the AIA Inspection, GEH determined that the ABWR DCD required additional updating to be consistent with its latest ABWR AIA. Therefore, in the applicant’s letters dated September 2, 2016, and its supplement dated November 23, 2016, GEH provided DCD Revision 6 markups based on information from TR NEDE-33875P, Revision 2 (ADAMS Accession No. ML16334A295 (proprietary)) that were necessary to update the ABWR DCD with information included in the latest ABWR AIA. The ABWR DCD, Revision 6 markups based on the update to TR NEDE-33875P, Revision 3, also identified additional editorial changes and additional fire protection key design features that protect the core cooling features. For example, the fire protection related changes include:

- Corrections to room and fire area numbers as well as adjusting rating locations of floor assemblies within DCD Tier 2, Figures 9A.4-3 through 9A.4-8;
- Addition of a new Inspections, tests, analyses, and acceptance criteria in DCD Tier 1, Table 2.15.10, to ensure the R/B steel trusses supporting the roof are encased with 5 psid (34.5 kPa), 3-hour fire rated material;
- Addition of new language under DCD Tier 2, Section 9A.2, to ensure the R/B steel trusses supporting the roof are encased with 5 psid (34.5 kPa), 3-hour fire rated material;
- Addition of a new key design feature stating cabling and ventilation routing is designed divisionally; and
- Addition of new constraint under DCD Tier 2, Section 19G.4.3-1, stating divisional power, instrumentation or control cabling routed through another space must be assessed under 10 CFR § 50.150.

These key design features, as described by GEH, ensure at least one complete train of heat removal equipment and necessary support systems (including cooling water, electrical power supply and distribution, and instrument and control) within the R/B are available to provide core cooling following the impact of a large commercial aircraft.

Based on the addition of the fire protection key design features listed above and the staff review of those additional design features, including those identified in the DCD Revision 6 markups, the staff finds the applicant’s description of the fire protection key design features for protecting core cooling equipment to be adequate and acceptable in accordance with 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*19.5(D).3.2 Plant Arrangement and Plant Structural Design Features*

In the DCD Tier 2, Section 19G.4.2, "Site Arrangement and Plant Structural Design," of the revised ABWR DCD, Revision 6, markups, the applicant stated that the ABWR plant design and arrangement of major structures as described in DCD Tier 2, Section 1.2, "General Plant Description," and Figure 1.2-1, "Site Plan," are key design features. The applicant also described key structural design features for aircraft impact in DCD Tier 2, Section 3H.6, "Summary of Key Structural Design Features," of the revised ABWR DCD, Revision 6, markups.

Specifically, the applicant stated that the AIA credited the arrangement and design of the building features to limit the location and effects of potential aircraft strikes on the R/B, RCCV and C/B. Sections 19.5(D).3.2.1–19.5(D).3.2.7 (below) of this supplemental FSER detail the staff's review of the design features and functional capabilities of those individual buildings to demonstrate that the acceptance criteria of 10 CFR § 50.150(a)(1) can be met.

*19.5(D).3.2.1 Location and Design of the Control Building*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the C/B to protect portions of the north wall of the R/B, and core cooling equipment.

In Item (1) of DCD Tier 2, Section 19G.4.2, of the ABWR DCD, Revision 6, markups, the applicant stated that the location and design of the C/B structure as described in DCD Tier 2, Section 3.8.4 and DCD Tier 2, Section 3H.2 are design features that protect portions of the R/B from the impact of a large commercial aircraft. The staff reviewed general arrangement drawings in DCD Tier 2, Figure 1.2-1, "Site Plan;" Figure 1.2-14 "Control and Service Building, Arrangement;" Figure 1.2-15 "Control and Service Building, Arrangement;" and Figure 1.2-22 "Control and Service Building, Arrangement Plan." The staff also reviewed DCD Tier 2, Section 3.8.4.1.2, "Control Building," and DCD Tier 2, Section 3H.2 "Control Building," and confirmed that the north wall of the R/B is protected by the shear walls of the C/B.

The applicant made additional changes in Item (1) of DCD Tier 2, Section 19G.4.2 to clarify that the C/B location, fixed with respect to other major structures, is defined in the TR NEDE-33875P, Revision 3, to ensure that credit of the C/B as an intervening structure is maintained. The staff reviewed the relevant drawings (DCD Tier 2, Figure 1.2-1; Figure 1.2-20, "Control and Service Building, Arrangement Plan;" Figure 1.2-21 "Control and Service Building, Arrangement Plan;" and Figure 1.2-22 "Control and Service Building, Arrangement Plan"), which show the relative relationship of the building locations among the C/B annex, C/B, and R/B structures. The staff further reviewed DCD Tier 2, Table 3-2, "Intervening Structures Credited in ABWR Aircraft Impact Assessment," and TR NEDE-33875P, Revision 3, Figure 3-1, "ABWR Site Plan - Location of Structures," which show the distance from the intervening structures to the shielded structure. The applicant screened the C/B as an intervening structure based on the criteria set in NEI 07-13, Revision 8 Section 3.2.2, “Screening Based on Intervening Structures.” The staff confirmed that the location of the relevant structures is fixed at the original DC stage. On this basis, the staff finds credit of the C/B as an intervening structure acceptable.

The applicant further added new Item (11) to the DCD Tier 2, Section 19G.4.2 clarifying that any permanent structure that penetrates the C/B roof will be sized to preclude a strike from the east and west direction. The applicant described in TR NEDE-33875P, Revision 3, that penetrations are not installed on the C/B roof without an AIA cognizant engineer review. The staff reviewed the DCD Tier 2, Figure 1.2-22, “Control and Service Building, Arrangement Plan,” and TR NEDE-33875P, Revision 3, Section 3.5, “Functional Success Criteria,” and found that permanent structure penetrations on the C/B roof in certain areas depend on AIA strike angles and roof penetration sizes. Therefore, the staff finds the design features and the controls established regarding permanent structure penetrations on the C/B roof acceptable.

Based on the above review, the staff finds that the applicant’s description of the C/B location, design, and its AIA analysis, as described in NEDE- 33875P, Revision 3, protects portions of the R/B from the impact of a large, commercial aircraft in accordance with the requirements of 10 CFR § 50.150(b). The staff also finds the applicant’s description of the design features and controls for permanent structure penetrations of the C/B roof to be acceptable and in accordance with the requirements of 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*19.5(D).3.2.2 Location and Design of the Turbine Building*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the T/B to protect the entire north wall of the C/B, portions of the north wall of the R/B, and core cooling equipment from the impact of a large, commercial aircraft.

In Item (2) of DCD Tier 2, Section 19G.4.2, of the revised ABWR DCD, Revision 6 markups, the applicant stated that the location and design of the T/B structure and layout as described in DCD Tier 1, Section 2.15.11 and Tier 2, Figures 1.2-24 through 1.2-31 are key design features that protect the entire north wall of the C/B and portions of the north wall of the R/B from the impact of a large, commercial aircraft. The staff reviewed general arrangement drawings in DCD Tier 2, Figures 1.2-1, and 1.2-24 through 1.2-31. The staff also reviewed DCD Tier 1, Section 2.15.11, “Turbine Building,” and finds that the T/B is designed such that damage to safety-related functions does not occur under seismic loads corresponding to the safe-shutdown ground acceleration. Review of these general arrangement drawings shows that entire north wall of the C/B and portions of the north wall of the R/B are protected by the T/B structure.

The applicant made additional changes in Item (2) to the DCD Tier 2, Section 19G.4.2, to clarify that the T/B location, fixed with respect to other major structures, is defined in TR NEDE-33875P, Revision 3, to ensure that credit of the T/B as an intervening structure is maintained. The staff reviewed the relevant drawings (DCD Tier 2, Figures 1.2-1, and 1.2-24 through 1.2-31), which show the relative relationship of the building locations among the T/B, C/B, and R/B structures. The staff further reviewed TR NEDE-33875P, Revision 3, Table 3-2, "Intervening Structures Credited in ABWR Aircraft Impact Assessment," and Figure 3-1, "ABWR Site Plan-Location of Structures," which show the distance from the intervening structures to shielded structure. The applicant screened the T/B as an intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2 as endorsed by RG 1.217, Revision 0. The staff confirmed that the location of the relevant structures is fixed at the original DC stage. On this basis, the staff finds the credit of the T/B as an intervening structure acceptable.

The applicant further added new Item (11) to DCD Tier 2, Section 3H.6 that included details of the T/B reinforced concrete exterior wall adjacent to the C/B. In TR NEDE-33875P, Revision 3, Table 5-1, "Key Structural Design Features in DCD Appendix 3H.6," the applicant described that this is an input to allow credit of the S/B wall as an intervening structure. The staff reviewed DCD Tier 2, Figures 1.2-1 and 1.2-25, "Turbine Building General Arrangement," and TR NEDE-33875P, Revision 3, Table 5-1, and finds them acceptable because the staff agreed with the applicant regarding how they screened the T/B as an intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2, as endorsed by RG 1.217,Revision 0.

Based on the above review, the staff finds the applicant's description, including location and design of the T/B structure and layout, as a key design feature for protecting the entire north wall of the C/B and portions of the north wall of the R/B from the impact of a large, commercial aircraft to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

#### *19.5(D).3.2.3 Location and Design of Reinforced Concrete Containment Vessel and Reactor Building Structure*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the RCCV and R/B structures to protect the safety systems located inside primary containment and the entire south wall of the C/B from the impact of a large, commercial aircraft. The applicant used the guidance provided in NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0, to perform detail structural analyses to determine the design of selected structures providing protections from the impact of a large commercial aircraft.

In Item (3) of DCD Tier 2, Section 19G.4.2, as revised in the ABWR DCD, Revision 6 markups, the applicant stated that the location and design of the R/B structure as described in DCD Tier 2, Sections 3.8.4, "Other Seismic Category I Structures," and

DCD Tier 2, Section 3.H1, “Reactor Building,” are the key design features protecting portions of the primary containment and the entire south wall of the C/B from the impact of a large, commercial aircraft. The applicant further described the protection provided from exterior walls, interior walls, intervening structures, and barriers on the large openings in the R/B exterior walls.

The staff reviewed DCD Tier 2, Section 3.8.4 and DCD Tier 2, Section 3.H1 and finds that the R/B and RCCV are reinforced concrete structures, below grade. Review of these general arrangement drawings (DCD Tier 2, Figures 1.2-1, and 1.2-4 through 1.2-12) shows that the entire south wall of the C/B is protected by the concrete shear walls of the R/B. Further, in DCD Tier 2, Section 19G.2, “Scope of Assessment,” of the revised ABWR DCD, Revision 6, markups, the applicant stated that the SFP and RCCV are not perforated in the event of an aircraft impact based on the assessment results; therefore, assessment of the damage to RCCV internal structures, systems and components (SSCs) and secondary impact is not required. In addition, the staff reviewed TR NEDE-33875P, Revision 3, Section 2.0, “Analysis Inputs,” Section 4.3, “Structural Assessment,” Table 4-2, “Summary of Material Specifications,” and Table 4-4, “Summary of Strengthening Measures. The staff finds that the applicant performed the assessment for the AIA using the methodology in NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0, strengthened measures for the interior and exterior walls based on results of the assessment; and designed external barriers as shown in DCD Tier 2, Figure 1.2-8, “Reactor Building, Arrangement Plan,” and DCD Tier 2, Figure 1.2-9, “Reactor Building, Arrangement Plan,” in combination with the external wall to protect the critical penetrations. Section 19.5(D).3.2.7 (below) of this supplemental FSER provides the technical evaluation of the adequacy of the reactor cavity shield blocks for protecting the drywell head from secondary impacts.

The applicant made additional changes in Item (3) to the DCD Tier 2, Section 19G.4.2 clarifying that TR NEDE-33875P, Revision 3, defines the R/B location, as fixed with respect to other major structures to ensure that credit of the R/B as an intervening structure is maintained. The staff reviewed the relevant drawings (DCD Tier 2, Figures 1.2-1, and 1.2-4 through 1.2-12), which show the relative relationship of the building locations among the T/B, C/B, and R/B structures. The staff further reviewed TR NEDE-33875P, Revision 3, Table 3-2 and Figure 3-1, which show the distance from the intervening structures to shielded structure. The applicant screened the R/B as an intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2 of, as endorsed by RG 1.217, Revision 0. The staff confirmed that the location of the relevant structures is fixed at the original DC stage. On these bases, the staff finds that crediting the R/B as an intervening structure is acceptable.

The applicant added new Item (10) to the DCD Tier 2, Section 19G.4.2 clarifying that the R/B crane will be parked at the north wall of the R/B when it is not used because doing so would significantly reduce the probability of the effect of secondary impact from falling crane components on the shield blocks that protects the drywell head from the impact of a large, commercial aircraft.

In Item (2) in DCD Tier 2, Sections 3H.6, of the revised ABWR DCD, Revision 6, markups, the applicant stated that the structural configuration of the RCCV within the R/B precludes direct strike on containment, and structural design of the RCCV ensures that the RCCV is not perforated. In addition, In DCD Tier 2, Section 19G.4.1, of the revised ABWR DCD, Revision 6, markups, the applicant described the RCCV as a key design feature that would protect the safety systems located inside primary containment from the impact of a large, commercial aircraft. The staff reviewed the description of key design features of RCCV in DCD Tier 2, Section 3.8 and DCD Tier 2, Section 3.H1. The staff also reviewed the description of RCCV material specifications in TR NEDE-33875P, Revision 3, Table 4-2, "Summary of Material Specifications," and Section 4.3, "Structural Assessment." The staff finds that the RCCV is not perforated in the event of an aircraft impact based on the assessment results; therefore, assessment of the damage to RCCV internal SSCs and secondary impact is not required. In Section 19.5(D).3.2.7 (below) of this supplemental FSER, provides the staff's independent review and assessment of the shield blocks protecting integrity of the drywall head from the secondary impacts.

The applicant made additional changes in Item (4) DCD Tier 2, Section 3H.6, of the revised ABWR DCD, Revision 6, markups clarifying that the interior partition walls are thickened and strengthened as shown in TR NEDE-33875P, Revision 3, to limit physical damage to interior partition walls from the impact of a large, commercial aircraft. The staff reviewed general arrangement drawings for the interior partition walls in DCD Tier 2, Figures 1.2-8 and 1.2-9. The staff also reviewed the description of thickened and strengthened internal partition walls in TR NEDE-33875P, Revision 3, Table 4-4. The staff finds that the interior partition walls are appropriately thickened and strengthened based on the results of the assessment in TR NEDE-33875P, Revision 3, and are therefore acceptable.

In Item (5) of DCD Tier 2, Sections 3H.6, of the revised ABWR DCD, Revision 6, markups, the applicant stated that the reinforced concrete sliding barriers with structural capacity equivalent to the surrounding wall are provided for the 6 large openings on 1F, as shown in DCD Tier 2, Figure 1.2-8, to limit physical damage to exterior walls. The staff reviewed DCD Tier 2, Figure 1.2-8 and TR NEDE-33875P, Revision 3, and finds that reinforced concrete sliding barriers in combination with the external wall are provided to protect the critical penetrations from the impact of a large, commercial aircraft and are therefore acceptable.

The applicant added new Item (12) to the DCD Tier 2, Section 3H.6 which clarifies that the R/B exterior walls on the east, west, and south sides are strengthened with enhanced reinforcement as described in TR NEDE-33875P, Revision 3. The staff reviewed the description of the enhanced reinforcement of the exterior wall on the east, west and south of the R/B in TR NEDE-33875P, Revision 3, Table 4-4. The staff finds that the east, west, and south sides are adequately strengthened with enhanced reinforcement based on the results of the assessment in TR NEDE-33875P, Revision 3, and are therefore acceptable.

Based on the above review, the staff finds the applicant's description of the location and design of the R/B and RCCV as the key structural design features for providing protection for maintaining core cooling to be adequate and acceptable, because the applicant described the physical protections and intervening structures to protect the primary containment (RCCV and drywell head) and the entire south wall of the C/B using the guidance of NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0, to perform detail structural analyses, and to determine the design of selected structures providing protections in accordance with 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the appropriate changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*19.5(D).3.2.4 Location and Design of Service Building Structure*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the S/B to protect the east wall of the C/B, and core cooling equipment.

In Item (6) of DCD Tier 2, Section 19G.4.2, of the revised ABWR DCD, Revision 6, markups, the applicant stated that the location and design of the S/B structure as described in DCD Tier 2, Section 3H.6 and Figures 1.2-20 through 1.2-22 are key design features that protect the east wall of the C/B from the impact of a large commercial aircraft. The staff reviewed general arrangement drawings in DCD Tier 2, Figures 1.2-1, and 1.2-14 through 1.2-22. The staff also reviewed DCD Tier 1, Section 2.15.14, "Service Building," and finds that the S/B is located adjacent to the C/B. Review of these general arrangement drawings show that the east wall of the C/B is protected by concrete shear wall of S/B.

The applicant made additional changes in Item (6) to the DCD Tier 2, Section 19G.4.2 that clarify that the S/B location, fixed with respect to other major structures, is defined in TR NEDE-33875P, Revision 3, to ensure that credit of the S/B as an intervening structure is maintained. The staff reviewed the relevant drawings (DCD Tier 2, Figures 1.2-1 and 1.2-24 through 1.2-22), which show relative relationship of the building locations among the S/B, C/B, and R/B structures. The staff further reviewed TR NEDE-33875P, Revision 3, Table 3-2, "Intervening Structures Credited in ABWR Aircraft Impact Assessment," and Figure 3-1, "ABWR Site Plan – Location of Structures," which show the distance from the intervening structures to shielded structure. The applicant screened the S/B as an intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2, of as endorsed by RG 1.217, Revision 0. The staff confirmed that the location of the relevant structures is fixed at the original DC stage. On these bases, the staff finds credit of the S/B as an intervening structure acceptable.

The applicant further added new Item (10) to DCD Tier 2, Section 3H.6 stating that the S/B exterior wall adjacent to the C/B is a reinforced concrete wall. In TR NEDE-33875P, Revision 3, Table 5-1 the applicant describes the S/B as an intervening structure. The staff reviewed DCD Tier 2, Figures 1.2-1 and 1.2-15, and TR NEDE-33875P, Revision 3, Table 5-1 and finds it acceptable, because the applicant screened the S/B as intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2 of as endorsed by RG 1.217, Revision 0.

Based on the above review, the staff finds the applicant's description, including location and design, of the S/B structure as key design features for protecting the east wall of the C/B from the impact of a large, commercial aircraft to be adequate and acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory

Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*19.5(D).3.2.5 Location and Design of Control Building Annex Structure*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the C/B annex building to protect the west wall of the C/B, and core cooling equipment.

In Item (7) of DCD Tier 2, Section 19G.4.2, of the revised ABWR DCD, Revision 6, markups, the applicant stated that the location and design of the C/B annex building structure as described in DCD Tier 2, Section 3H.6 and Figures 1.2-20 through 1.2-22 are key design features that protect the west wall of the C/B from the impact of a large, commercial aircraft. The staff reviewed general arrangement drawings in DCD Tier 2, Section 3H.6, Figures 1.2-1, and 1.2-20 through 1.2-22. The staff also reviewed DCD R Tier 1, Section 2.15.15, "Control Building Annex," and finds that the C/B annex is located adjacent to the C/B. Review of these general arrangement drawings show that west wall of the C/B is protected by the concrete shear walls of the C/B annex.

The applicant made additional changes in Item (7) to the DCD Tier 2, Section 19G.4.2 that clarify that the C/B annex location, fixed with respect to other major structures, is defined in the TR NEDE-33875P, Revision 3 to ensure that credit of the C/B Annex as an intervening structure is maintained. The staff reviewed the relevant drawings (DCD Figures 1.2-1, and 1.2-24 through 1.2-31), which show the relative relationship of the building locations among the C/B annex, C/B, and R/B structures. The staff further reviewed TR NEDE-33875P, Revision 3, Table 3-2 and Figure 3-1 which show the distance from the intervening structures to shielded structure. The applicant screened the C/B annex as an intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2, as endorsed by RG 1.217, Revision 0. The staff confirmed that the location of the relevant structures were fixed at the original DC stage. On these bases, the staff finds credit of the C/B annex as an intervening structure acceptable.

The applicant made additional changes in Item (9) of DCD Tier 2, Section 3H.6 that state that the C/B annex building exterior walls are made of reinforced concrete. The staff reviewed Figures 1.2-1 and 1.2-15 in DCD Tier 2, Revision 6, and TR NEDE-33875P, Revision 3, Table 5-1, and finds it acceptable, because the applicant screened the S/B as an intervening structure based on the criteria set in NEI 07-13, Revision 8, Section 3.2.2 of, as endorsed by RG 1.217, Revision 0.

Based on the above review, the staff finds the applicant's description, including location and design, of the C/B annex structure as key design features for protecting the west wall of the C/B from the impact of a large, commercial aircraft to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR § 50.150(b).

The applicant provided the necessary information from its letters dated September 2, 2016, and November 23, 2016, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6 markups. Therefore, Confirmatory Item 19.5-1 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

*19.5(D).3.2.6 The Seismic Gap between Reactor Building and Control Building*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the seismic gap between the R/B and C/B in protecting the C/B from shock effects from strikes on the R/B.

In Item (8) of DCD Tier 2, Section 19G.4.2 of the revised ABWR DCD, Revision 6, markups, the applicant stated that the seismic gap between the R/B and C/B described in DCD Section 3.8.5 is a key design feature in protecting the C/B from shock effects from strikes on the R/B. The staff reviewed DCD Tier 2, Section 3.8.5.1, and found that both R/B and C/B are supported by the reinforced concrete mat foundations, which are separated from each other by a gap of 2 meters (6 feet-6-3/4 inches) to minimize the structural interaction between the buildings. The staff also reviewed TR NEDE-33875P, Revision 3, Table 3-2 and Figure 3-1, which show the distance from the intervening structures to shielded structure. The staff confirmed that the seismic gap between R/B and C/B provided in the report is greater than 2 meters (6 feet-6-3/4 inches) and is therefore acceptable.

Based on the above review, the staff finds the applicant's description, including the seismic gap between the R/B and C/B as a key design feature for protecting the C/B from shock effects from strikes on the R/B to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR § 50.150(b).

*19.5(D).3.2.7 Shield Blocks Over Drywell Head*

The staff reviewed the ABWR DCD to ensure that the applicant performed a reasonably formulated assessment of the capability of the shield blocks to protect the integrity of the drywell head from the secondary impact of concrete debris, aircraft wreckage, and falling crane components resulting from the impact of a large commercial aircraft on the R/B. In Item (3) of DCD Tier 2, Section 3H.6, from the revised ABWR DCD, Revision 6, markups, the applicant stated that the shield blocks are configured to fully resist secondary impacts from concrete debris, aircraft wreckage and falling crane components to protect the integrity of drywell head. The applicant further stated that the shield blocks are placed over the drywell head in the reactor cavity between the pool girders as shown in DCD Tier 2, Figure 3H.1-23. The staff reviewed DCD Tier 2, Section 3H.1.3, "Description of the Containment and the Reactor Building," DCD Tier 2, Figure 3H.1-23, DCD Tier 2, Section 3H.6, and TR NEDE-33875P, Revision 3. As described in the ABWR DCD sections and in the TR NEDE-33875P, Revision 3, the shield blocks are configured to fully resist secondary impacts from concrete debris, aircraft wreckage and falling crane components to protect the integrity of drywell head.

Based on the above review, the staff finds the applicant's description of the shield blocks as the key structural design feature for providing physical protection of the integrity of the drywell head to be acceptable because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR § 50.150(b).

19.5(D).4 <u>Shock Damage</u>

In DCD Tier 2, Section 19G.2, the applicant stated that the analysis of aircraft impacts considers the effects of shock-induced vibrations on SSCs. In DCD Tier 2, Section

19G.4.1, “Primary Containment,” the applicant stated that safety-related components inside primary containment, including the reactor pressure vessel and associated ECCS piping are not adversely affected by shock-induced vibrations resulting from the impact of a large, commercial aircraft. In addition, DCD Tier 2, Section 19G.4.4, the applicant stated that all support systems were assessed for shock damage. Based on the applicant’s use of NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0, for its assessment scope that included shock-induced vibration, the staff finds that the applicant has performed a reasonably formulated shock analysis within the ABWR AIA.

19.5(D).5 Spent Fuel Pool Integrity

In DCD Tier 2, Section 19G.2, “Scope of Assessment,” of the revised ABWR DCD, Revision 6, markups, the applicant stated that the SFP and RCCV are not perforated, based on the assessment results, in the case of an aircraft impact; therefore, assessment of the damage to RCCV internal SSCs and secondary impact is not required. In Item (4) of DCD Tier 2, Section 19G.4.2, in the revised ABWR DCD, Revision 6, markups, the applicant stated the location and design, and its supporting structures as described in DCD Tier 2, Section 9.1 and Figure 1.2-12 are the key design features in protecting the SFP from the impact of a large commercial aircraft. However, the applicant did not describe whether an assessment was performed to ensure that required minimum water level in the SFP is maintained in the case of an aircraft impact. Therefore, on April 20, 2015, the staff issued RAI 19-6 (ADAMS Accession No. ML15110A122), requesting the applicant to confirm if an assessment was performed to ensure there is no leakage through the SFP liner below the required minimum technical specification water level of the pool. The applicant responded in a letter dated September 17, 2015 (ADAMS Accession No. ML15264A003), and submitted clarification in the DCD Tier 2, Section 19G.5, “Conclusions of Assessment,” “that the aircraft impact would not inhibit the ABWR’s core cooling capacity and SFP integrity based on the best estimate calculations.” Previously, as part of the DCD Revision 6, the applicant stated the following:

> [T]here are no AIA scenarios that would result in leakage from the SFP below the required minimum water level. The location and design of the SFP and its supporting structure preclude a direct hit from aircraft impact, therefore the pool liner is not perforated, and all piping attachments are configured such that they would not allow drain down below the minimum water level described in DCD Tier 2, Section 9.1.3.3, Safety Evaluation

The staff assessed the response and finds that the applicant adequately addressed this question since the aircraft impact would not inhibit the ABWR’s core cooling capability and spent SFP pool integrity based on best estimate calculations performed in accordance with NEI 07-13, Revision 8, as endorsed by RG 1.217, Revision 0. Therefore, the staff considers RAI 19-6 to be resolved and closed.

The applicant made additional changes to Item (1) in DCD Tier 2, Section 3H.6, of the revised ABWR DCD, Revision 6, markups. The applicant stated that (1) the structural configuration of the SFP within the R/B precludes a direct strike on the SFP, (2) the SFP is a reinforced concrete structure with a specified minimum thick ASTM A-240 Type 304L stainless steel liner, and (3) the SFP walls are strengthened as described in TR NEDE-33875P, Revision 3, to ensure the integrity of the SFP is maintained. The staff reviewed DCD Tier 2, Sections 9.1.2, “Spent-Fuel Storage,” and TR NEDE-33875P, Revision 3, and the staff confirmed that the SFP is a reinforced structure with a specified minimum thick stainless-steel liner and the SFP walls are strengthened.

Based on the above review, the staff finds the description of the key design features for ensuring SFP integrity to be acceptable, because the applicant adequately described the above design features and functional capabilities in accordance with 10 CFR § 50.150(b). The applicant provided the necessary information from its RAI response dated September 17, 2015, in the ABWR DCD, Revision 7, which incorporated the changes from the ABWR DCD, Revision 6, markups. Therefore, Confirmatory Item 19.5-2 from the staff advanced safety evaluation with no open items for the ABWR DC renewal is resolved and closed.

### *19.5.4 19.5(E) Conclusion*

The staff finds that the applicant has performed an AIA in ABWR DCD, Revision 7, that is reasonably formulated to identify design features and functional capabilities that show, with reduced use of operator action, that the acceptance criteria in 10 CFR § 52.59(a) and 10 CFR § 50.150(a)(1) are met.

The staff also finds that the applicant adequately described the key design features and functional capabilities identified and credited to meet 10 CFR § 50.150(b), including descriptions of how the key design features meet the acceptance criteria in 10 CFR § 50.150(a)(1); namely the facility can withstand the effects of a large commercial aircraft impact such that the reactor core remains cooled and SFP integrity is maintained. Therefore, the staff finds that the applicant meets the applicable requirements of 10 CFR § 50.150(b).

# 22 ENHANCEMENTS RESULTING FROM FUKUSHIMA NEAR TERM TASK FORCE RECOMMENDATIONS

This supplemental FSER Chapter, “Enhancements Resulting from Fukushima Near Term Task Force Recommendations,” documents the NRC staff’s evaluation or cites the specific staff supplemental FSER sections where the staff evaluated the GEH ABWR design enhancements in response to recommendations from the NTTF that the staff asked the applicant to address for renewal of the ABWR DC. The staff determined that the ABWR DC renewal applicant is not required to address the mitigation of beyond-design-basis events (MBDBE) rule (10 CFR § 50.155, “Mitigation of beyond-design-basis events”) that was published in the *Federal Register* on August 9, 2019 (84 FR 39684) and became effective September 9, 2019.[21] Prior to the implementation of the MBDBE rule, the staff had determined that the ABWR DC renewal applicant should address the following three NTTF topics: (1) mitigation strategies for beyond-design-basis external events (related to NTTF Recommendation 4.2), (2) spent fuel pool (SFP) instrumentation (related to NTTF Recommendation 7.1), and (3) emergency preparedness (EP) staffing and communications (related to NTTF Recommendation 9.3).

*Background:*

On March 11, 2011, a magnitude 9.0 earthquake struck off the coast of the Japanese island of Honshu. The earthquake resulted in a large tsunami that is estimated to have exceeded 14 meters (45 feet) in height, which inundated the Fukushima Dai-ichi Nuclear Power Plant site. The tsunami caused extensive damage to site facilities and resulted in a complete loss of all alternating current (ac) electrical power at 5 of the 6 units on the site.

In response to the events at the Fukushima Dai-ichi nuclear power plant resulting from the earthquake and tsunami in Japan, the NRC established the NTTF to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make improvements to its regulatory system, and to make recommendations to the Commission for policy directions. In July 2011, the NTTF identified 12 recommendations in a report, SECY-11-0093, “Near Term Report and Recommendations for Agency Actions Following the Events in Japan,” dated July 12, 2011 (ADAMS Accession No. ML11186A950). In SECY-11-0124, “Recommended Actions to be Taken Without Delay from the NTTF Report,” dated September 9, 2011 (ADAMS Accession No. ML11245A127), the staff submitted to the Commission for its consideration NTTF recommendations that could be and, in the staff’s judgment, should be, partially or entirely initiated without delay. In SECY-11-0124, the staff concluded that specific actions to address a subset of the NTTF recommendations would provide the greatest potential for improving safety in the near term. The staff also proposed three tiers of prioritization from the NTTF recommendations to the Commission in SECY-11-0137, “Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned,” dated October 3, 2011 (ADAMS Accession No. ML11269A204). The first tier consisted of those NTTF recommendations that the staff determined should be started without unnecessary delay and for which sufficient resource flexibility, including the availability of critical skill sets, existed. The second tier consisted of those NTTF recommendations that could not be initiated in the near term due to factors that included the need for further technical assessment and alignment, dependence on Tier 1 issues, and the availability of critical skill sets. The third

[21] The MBDBE final rule *Federal Register* notice also announced the public availability of the final regulatory guidance, Regulatory Guide (RG) 1.226, "Flexible Mitigation Strategies for Beyond-Design-Basis Events," Revision 0, and RG 1.227, "Wide-Range Spent Fuel Pool Level Instrumentation,” Revision 0, both issued in June 2019. Neither RG is applicable to the ABWR DC renewal.

tier consisted of NTTF recommendations that depended on the completion of near-term actions or needed additional study to support a regulatory action.

In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A111), the staff recommended that the Commission issue orders and requests for information under 10 CFR § 50.54(f) to power reactor licensees and stated that the staff would ask all combined license (COL) applicants to provide the requisite Tier 1 information addressed in the Commissions orders and the 10 CFR § 50.54(f) requests for information through the review process. The staff had determined that the following three Tier 1 NTTF recommendations should be addressed by the COL applicants at the time and the staff determined that the ABWR DC renewal applicant should consider design changes to address three Tier 1 NTTF recommendation topics for potential future ABWR DC COL applicants:

(1) Recommendation 4.2: Equipment covered under 10 CFR § 50.54(hh)(2) - Order licensees to provide reasonable protection for equipment currently provided pursuant to 10 CFR § 50.54(hh)(2) from the effects of design-basis external events, and to add equipment as needed to address multiunit events while other requirements are being revised and implemented.

(2) Recommendation 7.1: Spent fuel pool instrumentation - Order licensees to provide sufficient safety-related instrumentation, able to withstand design-basis natural phenomena, and to monitor SFP parameters (i.e., water level, temperature, and area radiation levels) from the control room.

(3) Recommendation 9.3: Emergency preparedness regulatory actions (staffing and communications.

In the July 20, 2012 letter, the NRC staff identified 28 items for GEH's consideration as part of their application to renew the ABWR DC. The applicant was requested by the staff in Item Nos. 26, 27 and 28 of that letter to identify design changes that would allow a COL applicant to address the Tier 1 Fukushima Recommendations 4.2,7.1, and 9.3, respectively. The staff addresses these requested changes below.

## 22.1 Mitigation Strategies for Beyond-Design-Basis External Events (NTTF Recommendation 4.2)

During the initial review of the application for ABWR DC renewal, the staff requested that GEH provide proposed changes to the ABWR design to address NTTF Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events. SECY-12-0025 states that the staff would request all COL applicants to provide the information addressed in the orders (i.e., EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735) (Mitigating Strategies Order), EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," dated March 12, 2012 (ADAMS Accession No. ML12054A694), and EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012 (ADAMS Accession No. ML12056A044) through the review process.

For mitigation strategies for beyond-design-basis external events, SECY-12-0025 outlines a three-phase approach. The initial phase involves the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling without both alternating current (AC) power and normal access to the ultimate heat sink. The transition phase involves providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from offsite. The final phase involves obtaining sufficient offsite resources to sustain those functions indefinitely.

In the staff's letter dated July 20, 2012, the staff requested that GEH address a compilation of design changes that the agency considered to be regulatory improvements or changes that could meet the criteria in 10 CFR § 52.59(b). In this letter the staff requested that GEH identify the design changes that would be incorporated into the DC renewal design control document (DCD) related to aspects of NTTF Recommendation 4.2, regarding mitigation strategies for beyond-design-basis external events, Item No. 26 of the letter. GEH responded to the staff in the letters described below addressing its proposed design changes to allow a potential COL applicant to meet requirements related to NTTF Recommendation 4.2.

On September 17, 2012 (ADAMS Accession No. ML12261A311), GEH responded to the staff's design suggestions by agreeing in the ABWR DCD, Revision 6, to incorporate the staff suggested design change items including Item No. 26 on mitigating strategies. In a letter dated September 9, 2015 (ADAMS Accession No. ML15254A042), GEH provided a detailed specific response with DCD markups to address Item No. 26 on mitigation strategies which was a follow-up from a public meeting on the issue held on May 7, 2015 (ADAMS Accession No. ML15162A613). The applicant provided details to address Attachment 2 of the Commission's Mitigating Strategies Order as requested by the staff.

In a public teleconference on March 17, 2016 (ADAMS Accession No. ML16124A049), the NRC staff requested that GEH clarify the ABWR response to a beyond-design-basis event with specific information items to be provided by the COL applicant that would also address the MBDBE proposed rule that was issued on November 13, 2015 (80 FR 70609). Therefore, in a letter dated April 29, 2016 (ADAMS Accession No. ML16120A032), GEH submitted its proposed resolution and supplemental information as requested by the staff during the March 17, 2016 public teleconference, including the ABWR DCD, Revision 6, markups and a proposed new Appendix 1D to the ABWR DCD that addresses the ABWR response to a beyond-design-basis event.. In a supplemental letter dated August 24, 2016 (ADAMS Accession No. ML16237A121), GEH provided additional updates to the previously submitted Appendix 1D and Enclosure 2 to the letter including the DCD markups associated with its supplemental response. GEH described how a licensee of an ABWR would use certain design features that are onsite, and what features are available when the plant transitions to using the equipment that could be brought in from offsite to maintain the plant in a safe condition.

As the NRC finalized the draft MBDBE final rule, it became clear that the staff would not require existing DCs including the ABWR, to address operational matters, such as those elements of the then draft proposed MBDBE rule. Therefore, the final rule would be consistent with the issue finality provision for the ABWR in 10 CFR § 52.63, "Finality of Standard Design Certifications." The staff describes this clarification for DCs in more detail in the regulatory analysis of the then proposed rule (ADAMS Accession No. ML15266A133).

Therefore, in a letter dated December 6, 2016 (ADAMS Accession No. ML16341A812), regarding the latest public information related to the draft MBDBE final rule and considering that no MBDBE rule requirements would be relevant to applicants for a standard DC (or a DC renewal, as in the case of the ABWR application), GEH stated that it planned to submit a revised response addressing Item No. 26 by the end of January 2017. The revised response would provide a complete description of the changes to the ABWR DCD that would remove references to NTTF Recommendation 4.2 mitigating strategies (e.g., Appendix 1D). In its followup response dated January 23, 2017 (ADAMS Accession No. ML17025A386), GEH submitted its final proposal to remove references to NTTF Recommendation 4.2 mitigating strategies, and therefore remove any reference or applicability related to the MBDBE rule for the ABWR DC renewal (e.g., Appendix 1D of the ABWR DCD). In addition, to the extent that certain design features were proposed in response to Item No. 26, GEH identified in its revised response which of those would be retained for NRC review as voluntary design changes in the renewal application (e.g., external connections for power and water; enhanced systems capability for residual heat removal (RHR) and reactor core isolation cooling (RCIC). Therefore, the staff reviewed these design enhancements as separate design elements not required or related to the MBDBE rule, in separate staff SERs as follows:

- DCD Tier 1 and 2, Chapter 5, “Reactor Coolant System and Connected Systems.”
  Supplemental SER Section 5.4.7.1.1.10, “ACIWA,” provides the staff’s evaluation of the DCD design amendment proposed by GEH for the addition of a redundant alternating current independent water addition (ACIWA) capability to the RHR Loop B and to provide clarity on the wetwell spray and SFP makeup capabilities of the ACIWA system.
- DCD Tier 2, Chapter 5, “Reactor Coolant System and Connected Systems.”
  Supplemental SER Section 5.4.7, “Residual Heat Removal System,” provides the staff’s evaluation of the DCD design amendment proposed by GEH for a redundant ACIWA mode to the RHR Loop B.
- DCD Tier 2, Chapter 7, “Instrumentation and Control Systems.”
  Supplemental SER Section 7.4.1.4.4, “Shutdown Panel,” provides the staff’s evaluation of the DCD design amendment proposed by GEH for additional controls and indications on the ABWR Remote Shutdown Panel.
- DCD Tier 2, Chapter 8, “Electric Power.”
  Supplemental SER Section 8.3.4.4, “Isolation Between Class 1E Buses and Loads Designated as Non-Class 1E,” provides the staff’s evaluation of the DCD design amendment proposed by GEH for a capability to provide electrical power to safety-related loads from an external non-safety power source.
- DCD Tier 2, Chapter 16, “Technical Specifications.”
  Supplemental SER Section 16, “Technical Specifications,” provides the staff’s evaluation of the DCD design amendment proposed by GEH for addition of ACIWA mode to RHR Loop B (currently available for RHR Loop C), affecting TS 3.5.1, “ECCS-Operating,” and TS 3.6.2.4, “RHR Containment Spray;” and, additional controls and indications on the ABWR Remote Shutdown Panel.

The ABWR design enhancements GEH provided in the ABWR DCD, Revision 7, may provide a potential COL applicant the means for meeting the MBDBE rule requirements for mitigating strategies.

## 22.2 Reliable Spent Fuel Pool Instrumentation (NTTF Recommendation 7.1)

In this ABWR supplemental FSER section, the staff evaluates the design changes proposed by GEH to address Fukushima NTTF Recommendation 7.1 regarding SFP reliable level instrumentation. These proposed design changes affect the following ABWR DCD Sections:

- DCD Tier 2, Chapter 3," Design of Structures, Components, Equipment and Systems." Supplemental FSER Section 3.2.3, "Safety Classifications," provides a pointer to this Supplemental FSER Section for the staff evaluation of the design changes made by GEH for the SFP level instrumentation to address the NTTF recommendation for reliable SFP instrumentation.
- DCD Tier 2, Chapter 7, "Instrumentation and Control Systems."
  Supplemental FSER Section 7.5.2.1, "Post Accident Monitoring System," provides a pointer to this Supplemental FSER Section for the staff evaluation of the design changes made by GEH for the SFP level instrumentation to address the NTTF recommendation for reliable SFP instrumentation.
- DCD Tier 2, Chapter 9, "Auxiliary Systems."
  Supplemental SER Section 9.1.3, "Fuel Pool Cooling and Cleanup System," provides a pointer to this Supplemental FSER Section for the staff evaluation of the design changes made by GEH for the SFP level instrumentation to address the NTTF recommendation for reliable SFP instrumentation.

In responding to and managing the damage caused by the event at Fukushima, those plant operators lacked, among other things, reliable instrumentation to determine the water level in the SFPs on the site. This lack, combined with the operators' inability to visually observe the SFPs because of the conditions in the plant, raised concerns that at least one pool may have boiled dry—resulting in fuel damage—and highlighted the need for reliable SFP instrumentation.

Although the likelihood of a catastrophic event affecting nuclear power plants and the associated SFPs in the United States remains very low, beyond-design-basis external events could challenge the ability of existing SFP instrumentation to provide emergency responders with reliable information on the condition of SFPs. A reliable and available indicator is essential to ensure that plant personnel can effectively prioritize emergency actions.

In SECY-12-0025, the NRC staff states that for DCs and COL applications submitted under 10 CFR Part 52 that are currently under active staff review, the staff plans to ensure that the Fukushima NTTF recommendations approved by the Commission are addressed before certification or licensing.

The Japan Lessons-Learned Project Directorate (JLD)-Interim Staff Guidance (ISG)-2012-03 Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation" (ADAMS Accession No. ML12221A339), endorses with exceptions and clarifications the methodologies described in the Nuclear Energy Institute (NEI) industry guidance document NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,'" Revision 1, (ADAMS Accession No. ML122400399), and provides an acceptable approach for satisfying the applicable requirements.

### *22.2.1 Regulatory Criteria*

The applicant proposed safety-related SFP level instrument design changes to the GEH ABWR DCD to provide reliable SFP level indication from the normal range to a level down to one meter below the top of active fuel. In addition, the instrument can be powered from an independent power source and power interruption will not impact the design accuracy. Therefore, in accordance with 10 CFR § 52.59(c), this design change is an “amendment,” as this term is defined in Chapter 1 of this SER supplement and will correspondingly be evaluated using the regulations in effect at renewal.

The applicant included a COL Information Item under DCD Section 7.5.3, describing the maintenance, implementation and training for these safety-related SFP level instruments. The applicant also added a DCD Section 7.5.4, listing the pertinent references used to implement the Commission Order regarding reliable SFP instrumentation.

### *22.2.2 Summary of Technical Information*

By letter dated September 25, 2014 (ADAMS Accession No. ML14267A352), the NRC staff in RAI Question 01.05-1 requested that GEH address the design-related aspects of Fukushima NTTF Recommendation 7.1 regarding enhanced spent fuel instrumentation as outlined in Attachment 2 of Order EA-12-051. The applicant responded to the staff’s RAI in letters dated November 6, 2014 (ADAMS Accession No. ML14310A567), June 18, 2014 (ADAMS Accession No. ML15170A045), and August 25, 2015 (ADAMS Accession No. ML15237A192). As part of the RAI response, the applicant added SFP level instruments that comply with applicable guidance. This change resulted in changes as reflected in the ABWR DCD, Revision 7, to the following Sections:

- DCD Tier 1, Section 2.6.2, Figure 2.6.2 and Table 2.6.2
- DCD Tier 2, Chapter 1, Tables 1.8-21 and 1.8-22
- DCD Tier 2, Chapter 3, Table 3.2-1
- DCD Tier 2, Chapter 7, Sections 7.5.2.1, 7.5.3 and 7.5.4
- DCD Tier 2, Chapter 9, Sections 9.1.3.2 and 9.1.7
- DCD Tier 2, Chapter 21, Figure 9.1-1

### *22.2.3 Technical Evaluation*

Commission Order EA-12-051 requires a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel. NEI 12-02 refers to these monitoring levels as Level 1, Level 2 and Level 3, respectively:

(1) level that is adequate to support operation of the normal fuel pool cooling system,

(2) level that is adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and

(3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

In the applicant's response to RAI 01.05-1, GEH stated that the ABWR design departs from the guidance of NEI 12-02 in the choice of water level nomenclature.  In accordance with human factors engineering principles, the ABWR SFP and RPV water level nomenclature have been made as consistent as possible.  Thus, the ABWR DCD designates SFP Level 3 as slightly below normal water level (EA-12-051 item (1) or NEI 12-02 Level 1), and Level 1 as above the top of active fuel (EA-12-051 item (3) or NEI 12-02 Level 3).

The staff evaluated the applicant's response and found that the proposed departure from the guidance was acceptable.  Changing the nomenclature of the levels has no adverse impact on safety, as long as all three levels are monitored and alarmed.  During a public meeting with the applicant on August 13, 2015 (ADAMS Accession No. ML15230A204), the staff identified that the applicant had introduced an additional departure from the guidance, without providing adequate justification for how the alternative meets the SFP instrumentation requirements.  The markups of DCD Tier 2, Section 9.1.3.2, (that were part of the RAI response) showed that the alarm setpoint for GEH Level 1 (lowest level) would be at the top of the active fuel.  This setpoint is not consistent with NEI 12-02, Level 3 (lowest level) which corresponds to the highest point of any fuel rack seated in the SFP.

On August 20, 2015, the applicant submitted a revised response to RAI 01.05-1 based on feedback provided during the August 13, 2015 public meeting.  In the revised response, GEH updated the lowest level alarm to be the top of the fuel assembly bail handle in ABWR DCD Tier 2, Subsection 9.1.3.2.

The staff finds the DCD changes meet the guidance in NEI 12-02 and therefore are acceptable.  The staff has confirmed that ABWR DCD, Revision 7 incorporated the markups provided in RAI 01.05-1.  Order EA-12-051 also requires the SFP instrumentation to include several design features.  The discussion below describes the design features (the key words are underlined).  All other aspects of RAI 01.05-1 have also been resolved by the applicant.

<u>Instrument:</u>

Commission Order EA-12-051, Attachment 2, Section 1.1 states that the SFP level instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel.  The backup instrument channel may be fixed or portable.  Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor the SFP water level under conditions that restrict direct personnel access to the pool, such as partial structural damage, high radiation levels, or high heat and humidity from a boiling pool.

The applicant's response to RAI 01.05-1 states that the instrumentation will consist of two safety related, permanent and fixed instrument channels.  DCD Revision 6, Tier 2, Section 7.5.2.1 states that the instruments are designed to remain reliable considering normal operational, event and post-event conditions.

The staff evaluated the applicant's changes to the DCD description and determined that crediting two safety-related permanently installed instruments as primary and backup channels conforms with the design features identified in staff guidance (i.e., JLD-ISG-2012-03).  Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051.  Therefore, this part of RAI 01.05-01 is resolved.  The staff has confirmed that Revision 7 of the DCD includes the markups provided in the response to RAI 01.05-1.

Arrangement:

Commission Order EA-12-051, Attachment 2, Section 1.2, states that the SFP level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the SFP. This protection may be provided by locating the primary instrument channel and fixed portions of the backup instrument channel, if applicable, to maintain instrument channel separation within the SFP area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the SFP structure.

In the applicant's response to RAI 01.05-1, GEH proposed markups to DCD Tier 2, Section 9.1.3.2 states that the SFP level instrument channels will be arranged in a manner that provides reasonable protection of the level indication function against external missiles. This protection will be provided by maintaining instrument channel separation within the SFP area and will utilize inherent shielding from missiles provided by the existing SFP structure. The channel separation guidance in NEI 12-02, Revision 1, Section 3.2, will be considered in determining sensor locations.

The staff evaluated the applicant's instrument location description provided in the ABWR DCD and determined that the applicant's changes conform to staff guidance (i.e., JLD-ISG-2012-03). Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-01 is resolved. The staff has confirmed that Revision 7 of the DCD includes the DCD markups provided in response to RAI 01.05-1.

Mounting:

Commission Order EA-12-051, Attachment 2, Section 1.3 states that the installed instrument channel equipment within the SFP shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the SFP structure. DCD Tier 2, Table 3.2-1, "Classification Summary," identifies that the SFP wide range level instrumentation is classified as a Seismic Category I component. The staff evaluated the applicant's ABWR DCD description and the equipment description included in the response to RAI 01.05-1 and determined that the applicant's changes conforms to staff guidance (i.e., JLD-ISG-2012-03). Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-1 is resolved.

Qualification:

Commission Order EA-12-051, Attachment 2, Section 1.4 states, in part, that the primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the SFP water at saturation conditions for an extended period.

The applicant's response to RAI 01.05-1 states that the instrument channels depart from the guidance of NEI 12-02 (Revision 1) in that the instrument reliability does not need to consider post-accident conditions of borated water. Boiling-water reactor (BWR) SFPs do not use borated water. DCD Tier 2, Section 7.5.2.1 states that the augmented quality assurance process will ensure that the level instrumentation will be operational at conditions (temperature, humidity and radiation levels) in the vicinity of the SFP and the area of use considering normal operational, event and post-event conditions for no fewer than seven days post-event or until off-site resources can be deployed by the mitigating strategies.

The staff evaluated the applicant's departure from the approved guidance and found it acceptable. Because borated water is not used in the BWR SFP, the instruments are not expected to be exposed to post-accident borated water conditions. The staff reviewed the applicant's response and the information in the DCD and determined that the instruments will be designed to remain operational during all other post-accident anticipated conditions of temperature, humidity and radiation levels and these capabilities will be demonstrated in accordance with the guidance in JLD-ISG-2012-03. Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-01 is resolved. The staff has confirmed that Revision 7 of the DCD includes the DCD markups provided in the response to RAI 01.05-1.

<u>Independence:</u>

Commission Order EA-12-051, Attachment 2, Section 1.5 states that the primary instrument channel shall be independent of the backup instrument channel. DCD Tier 2, Section 7.5.2.1 states that the instrument channels are powered from two independent Class 1E batteries. DCD Tier 2, Section 9.1.3.2 identifies the level transmitters as safety-related independent instruments.

The staff reviewed the applicant's response to RAI 01.05-01 along with the ABWR DCD changes. The staff verified that the physical separation of the channels will be sufficient to establish physical and electrical independence. Accordingly, the staff finds that this feature conforms to the guidance in JLD-ISG-2012-03. Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-01 is resolved. The staff has confirmed that Revision 7 of the DCD includes the DCD markups provided in the response to RAI 01.05-1.

<u>Power Sources:</u>

Commission Order EA-12-051, Attachment 2, Section 1.6 states, in part, that permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant alternating current (ac) and direct current (dc) power distribution systems, such as portable generators or replaceable batteries.

In the applicant's response to RAI 01.05-01, GEH proposed changes to DCD Tier 2, Section 7.5.2.1 to indicate that the level instrument channels will be powered by Class 1E batteries. In addition, the instruments will have the capability of being powered from an independent power source.

The staff identified that the level instrument channels are powered by separate Class 1E DC batteries capable of powering the instruments. The applicant designed the system with the capability of using an alternate power source to power the level instrumentation. Based on the evaluation of the system description provided in the DCD, the staff concludes that these design features conform to the guidance in JLD-ISG-2012-03. Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-01 is resolved. The staff has confirmed that Revision 7 of the DCD includes the DCD markups provided in the response to RAI 01.05-1.

Accuracy:

Commission Order EA-12-051, Attachment 3, Section 1.4 states that the instrument shall maintain its designed accuracy following a power interruption or change in power source without recalibration.

In the applicant's response to RAI 01.05-01, GEH proposed changes to DCD Tier 2, Section 7.5.2.1 to clarify that an interruption of power to the instruments will not impact the design accuracy of the instruments or require recalibration of the equipment.

The staff evaluated the applicant's response to RAI 01.05-01 and its proposed changes to the DCD instrument description and determined that the applicant conforms to staff guidance (i.e., JLD-ISG-2012-03). Because the applicant conforms to the staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-01 is resolved. The staff has confirmed that ABWR DCD, Revision 7, includes the DCD markups provided in the response to RAI 01.05-1.

Testing:

Commission Order EA-12-051, Attachment 2, Section 1.8 states that the instrument channel design shall provide for routine testing and calibration. The DCD described the level channels as permanently installed safety-related instrumentation. The COL information item in DCD Section 7.5.3.1, states that the COL applicant will provide information to ensure that SFP instrumentation shall be maintained to be available in accordance with the requirements of Commission Order EA-12-051, Attachment 2 and that the permanently installed instrument channels are normally used to monitor the SFP level and will be subject to routine testing and calibration in accordance with plant procedures. Therefore, this part of the RAI 01.05-1 is resolved.

Display:

Commission Order EA-12-051, Attachment 2, Section 1.9 states that trained personnel shall be able to monitor the SFP water level from the control room, the alternate shutdown panel, or another appropriate and accessible location. The display shall provide on-demand or continuous indication of SFP water level.

In the applicant's response to RAI 01.05-01, GEH proposed changes to DCD Tier 2, Section 9.1.3.2 to indicate that SFP water level can be monitored from the control room, the Remote Shutdown Panels, or other appropriate location accessible post-accident. Tier 1, Section 2.6.2, was revised to include ITAAC 2.6.2 Item 7 which requires verification that the safety-related level instruments provide level indication in the main control room and an alternate location.

The staff reviewed the applicant's changes to the system description in DCD Tier 1 and 2. The location of the level indication display, as installed, will be verified through testing, which will be confirmed through ITAAC 2.6.2-7, as discussed above. The staff finds that the applicant conforms to staff guidance (i.e., JLD-ISG-2012-03). Because the applicant conforms to staff guidance, the staff finds the applicant complies with Commission Order EA-12-051. Therefore, this part of the RAI 01.05-01 is resolved. The staff has confirmed that Revision 7 of the DCD includes the DCD markups provided in RAI 01.05-1.

Programs:

Commission Order EA-12-051, Attachment 2, Section 2 states that the SFP instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program, procedures, and a testing and calibration program. Personnel shall be trained in the use of the primary and backup instrument channels, provision of alternate power to each channel and testing and calibration of each channel. Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent SFP instrument channels. Processes shall be established and maintained for scheduling and implementing testing and calibration of the primary and backup SFP level instrument channels sufficient to maintain them at the design accuracy.

In DCD Tier 2, Section 7.5.3, "COL License Information," the applicant in COL Information Item 7.5.3.1, "Spent Fuel Pool Level Instruments," states:

> *In Commission Order EA-12-051, Attachment 2, Section 2 (Reference 7.5-3) states that the SFP instrumentation shall be maintained to be available and reliable through the appropriate development and implementation of a training program. Personnel shall be trained in the use and maintenance (including test and calibration), and in the procedures for providing alternate power to the level instrument channels.*

The staff finds that the COL Information Item 7.5.3.1, conforms to the guidance in JLD-ISG-2012-03, which addresses the development of procedures for testing and calibration of the primary and backup SFP level instrument channel, and therefore complies with Commission Order EA-12-051. The staff has also determined that the existing commitments in Final Safety Analysis Report Section 13.5, "Plant Procedures," already cover the procedures for the use of the safety-related permanently installed SFP level instrumentation. Therefore, no new commitment is needed for the development of these procedures. Accordingly, this part of the RAI 01.05-01 is resolved.

Based on the discussion presented above, the staff finds that all parts of the staff's concerns identified in the response to RAI 0.05-01 have been addressed and found acceptable, therefore RAI 01.05-01 is considered resolved and closed in its entirety.

ITAAC:

DCD Revision 6, Tier 1, Section 2.6.2 discusses a new ITAAC in Table 2.6.2 (as shown below), to ensure that the SFP level instrumentation will be designed and installed as described in Tier 1, Section 2.6.2.

**Tier 1, Table 2.6.2 Fuel Pool Cooling and Cleanup System**

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 7. The safety-related displays provided for the FPC System spent fuel pool wide range water level are as described in Section 2.6.2. | 7. Inspections will be performed of the safety-related FPC system displays in both the main control room and at an alternate location. | 7. Displays exist or can be retrieved in both the main control room and an alternate location. |

As discussed above (in Display supplemental Section), the staff finds that the new Fuel Pool Cooling and Clean-up System ITAAC acceptance criteria will confirm that the installed level instrumentation meets the design functions specified in Tier 1, Section 2.6.2. Therefore, the staff finds that the new ITAAC is acceptable because it meets the requirements of 10 CFR § 52.47(b)(1) with respect to the fuel pool cooling and cleanup system.

COL Information Item:

ABWR DCD, Revision 7, includes a COL Information Item in DCD Section 7.5.3.1, which instructs the COL applicants to develop and implement a training program for the use and maintenance of the SFP level instrumentation. As discussed above (in the *Testing* and *Programs* Section of this SER), the staff finds that the COL Information Item conforms to the guidance in JLD-ISG-2012-03.

### *22.2.4 Conclusion*

Order EA-12-051 required a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the pool water level conditions by trained personnel. In addition, the Order required that SFP level instrumentation include several design features (e.g., redundant instruments, separation and environment qualification). Based on the evaluation discussed above, the staff concludes that the applicant's design conforms with the guidance in JLD-ISG-2012-03, where appropriate, and therefore, is acceptable. As a result, the staff finds these instruments to be reliable, able to withstand design-basis natural phenomena, and capable of monitoring key SFP level conditions that address NTTF Recommendation 7.1 and meet the relevant requirements of the March 12, 2012, Order EA-12-051. The regulation in 10 CFR § 50.155(e), "Spent fuel pool monitoring," makes the requirements of NRC Order EA-12-051 generically applicable for operating plants under 10 CFR Part 50 and COL license holders under 10 CFR Part 52 for which the Commission has made the finding under 10 CFR § 52.103(g). The MBDBE rule is not applicable or required for DC applicants, however the design change enhancements provided by GEH to address NTTF Recommendation 7.1 regarding SFP reliable level instrumentation for the ABWR DC renewal, provided in the ABWR DCD, Revision 7, may provide a potential COL applicant the means for meeting 10 CFR § 50.155(e).

## 22.3 Emergency Preparedness (NTTF Recommendation 9.3)

The objective of EP is to ensure that the capability exists for a licensee (or will exist for a COL applicant) to implement measures that mitigate the consequences of a radiological emergency and to provide for protective actions of the public. The accident at Fukushima highlighted the need to determine the staffing needed to respond to a multi-unit event. Additionally, there is a need to ensure that the communication equipment relied on has adequate power to coordinate the response to an event during an extended loss of ac power. Requiring these staffing and communication capabilities were part of NTTF Recommendation 9.3.

In ABWR DCD, Revision 7, which incorporated DCD markups included in responses to RAIs, GEH made changes to the ABWR design to address various aspects of EP, in support of its ABWR DC application. In finalizing the MBDBE rule the enhanced EP capability related to Fukushima NTTF Recommendation 9.3 was removed as a requirement in the rule prior to the final rule affirmation by the Commission. Staffing and communications were removed from the draft final MBDBE rule by the Commission in its January 24, 2019 SRM-M190124A (ADAMS ML19023A038). The applicant was informed of this subsequently and prior to the completion of this supplemental FSER in Phase B of the review and GEH declined the option to revise its ABWR DCD to remove the EP enhancements related to NTTF Recommendation 9.3 that would be applicable to a potential COL applicant.

The staff reviewed these ABWR DCD design enhancements in a separate staff supplemental FSER Section as follows:

DCD Tier 2, Chapter 13, “Conduct of Operations.”

> Supplemental SER Section 13.3, “Emergency Planning,” provides the staff evaluation of the ABWR DCD design modifications to (1) ensure that site-specific radiological protection for the technical support center (TSC) will be verified at the combined license (COL) application stage, consistent with the applicable TSC habitability guidance, and (2) provide for an assessment of staffing and communications capabilities to respond to a beyond-design-basis-event, pursuant to certain NRC actions arising out of the Fukushima NTTF Recommendation 9.3.

# APPENDIX A
# CHRONOLOGY OF THE ABWR DC LICENSE RENEWAL

This appendix contains a chronological listing of routine licensing correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC) regarding the review of the ABWR DC License Renewal under Docket No. 052-000045.

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 12/07/2010 | ML110040176 | ABWR DC Renewal Application | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/07/2010 | ML110040323 | E-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 5 | DCD | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 2/14/2011 | ML110330550 | Federal Register Notice - Acceptance Review In The GE-Hitachi Nuclear Energy Design Certification Renewal Application For The U.S. Advanced Boiling Water Reactor. | Federal Register Notice | NRC/NRO/ DNRL/BWR | | 5200045 |
| 2/14/2011 | ML110330549 | Acceptance Review In The GE-Hitachi Nuclear Energy Design Certification Renewal Application For The U.S. Advanced Boiling Water Reactor. | Letter | NRC/NRO/ DNRL | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 2/15/2011 | ML110450004 | Acceptance Review Results for the GE-Hitachi Advance Boiling Water Reactor (GEH ABWR) Design Certification Renewal Application (TAC NO. RP0509). | Memoranda | NRC/NRO/ DSRA/SBPA | | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 2/15/2011 | ML110450009 | Resource Plan for GEH ABWR DC Renewal. | Spreadsheet File | NRC/NRO | | 5200045 |
| 3/4/2011 | ML110340360 | GE Hitachi Nuclear Energy Boiling Water Reactor Design Certification Renewal Application. | Letter | NRC/NRO/ DNRL | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 3/4/2011 | ML110620195 | 03/24/2011 Notice of Forthcoming Advanced Boiling Water Reactor Design Certification Renewal Application Meeting To Discuss GE-Hitachi Nuclear Energy Application. | Meeting Notice | NRC/NRO/ DNRL/BWR | NRC/NRO/DNRL/BWR | 5200045 |
| 3/24/2011 | ML110820621 | 03/24/2011 GE-Hitachi Slide Presentation for NRC Public Meeting on ABWR Design Certification Renewal - Draft Review Guidance. | Meeting Briefing Package/Handouts | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/15/2011 | ML110450009 | Resource Plan for GEH ABWR DC Renewal. | Spreadsheet File | NRC/NRO | | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 4/4/2011 | ML11126A173 | ABWR Design Certification, Design Control Document - Revision 4. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A099 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A134 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A136 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 00-Introduction | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A101 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 01-Introduction and General Description of Plant | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A137 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 01-Introduction and General Description of Plant | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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|---|---|---|---|---|---|---|
| 4/4/2011 | ML11126A102 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 02 Site Characteristics | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A138 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 02 Site Characteristics | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A103 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 03-Design of Structures, Components, Equipment, and Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A139 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 03-Design of Structures, Components, Equipment, and Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A104 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 04-Reactor | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A140 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 04-Reactor | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 4/4/2011 | ML11126A141 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 05-Reactor Coolant System and Connected Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A105 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 05-Reactor Coolant System and Connected Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A142 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 06-Engineered Safety Features | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A106 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 06-Engineered Safety Features | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A107 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 07-Instrumentation and Control Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A143 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 07-Instrumentation and Control Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A144 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 08-Electric Power | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 4/4/2011 | ML11126A108 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 08-Electric Power | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A109 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 09-Auxiliary Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A145 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 09-Auxiliary Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A146 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 10-Steam and Power Conversion System | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A110 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 10-Steam and Power Conversion System | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A147 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 11-Radioactive Waste Management | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A111 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 11-Radioactive Waste Management | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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|---|---|---|---|---|---|---|
| 4/4/2011 | ML11126A150 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 12-Radiation Protection | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A113 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 12-Radiation Protection | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A114 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 13-Conduct of Operations | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A151 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 13-Conduct of Operations | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A152 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 14-Initial Test Program | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A116 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 14-Initial Test Program | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A153 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 15-Accident and Analysis | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A117 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 15-Accident and Analysis | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 4/4/2011 | ML11126A154 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 16-Technical Specifications | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A118 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 16-Technical Specifications | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A155 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 17-Quality Assurance | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A119 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 17-Quality Assurance | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A156 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 18-Human Factors Engineering | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A120 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 18-Human Factors Engineering | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A121 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 19-Response to Severe Accident Policy Statement | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A157 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 19-Response to Severe Accident Policy Statement | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 4/4/2011 | ML11126A122 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 20- Question and Response Guide | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A158 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 20- Question and Response Guide | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A123 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21- Volume 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A159 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21- Volume 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A160 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21- Volume 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A124 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21- Volume 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A161 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21- Volume 3 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A125 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21- Volume 3 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 4/4/2011 | ML11126A162 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21-Volume 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A126 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21-Volume 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A127 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21-Volume 5 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A163 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21-Volume 5 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A164 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21-Volume 6 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A128 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Chapter 21-Volume 6 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A135 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Tier_1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/4/2011 | ML11126A100 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 4 - Tier_1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 4/18/2011 | ML111020492 | 03/24/2011-Summary of Meeting Between the USNRC Staff and GEH To Discuss GEH's ABWR Design Certification Renewal Application. | Meeting Summary | NRC/NRO/DNRL/BWR | NRC/NRO/DNRL/BWR | 5200045 |
| 4/18/2011 | ML111020602 | 03/24/2011 Attendance List for Meeting of USNRC Staff with GEH To Discuss ABWR Design Certification Renewal Application. | No Document Type Applies | NRC/NRO/DNRL/BWR | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 5/5/2011 | ML11131A032 | Transmittal of Information Related to Design Control Document Section 13.6, Physical Security, in Support of ABWR Standard Plant Design Certification Renewal Application. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/10/2011 | ML11192A261 | Memo, Amendments to the Advanced Boiling Water Reactor (ABWR) Standard Design that the Nuclear Regulatory Commission Recommends General Electric - Hitachi (GEG) Consider for Inclusion in the Application to Renew the ABWR Standard Design Certification. | Memoranda-Non-Public | NRC/NRO/DE/ICE2 | NRC/NRO/DNRL/BWR | 5200045 |
| 9/6/2011 | ML112490242 | Response to Request for a Review for Additional Amendments to GE-Hitachi ABWR Design Certification Renewal Application. | Memoranda Non-Public | NRC/NRO/DCIP/CTSB | NRC/NRO/DNRL/BWR | 5200045 |
| 9/23/2011 | ML112650080 | Request for Additional Amendments to General Electric-Hitachi ABWR Design Certification Renewal Application. | Memoranda Non-Public | NRC/NRO/DSRA/SPCV | NRC/NRO/DNRL/BWR | 5200045 |

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| 11/23/2011 | ML113260553 | Memo, Application of the State-Of-The-Art Human Factors Principles in The Control Room Design. | Memoranda | NRC/NRO/DCIP/COLP | NRC/NRO/DNRL/NARP | 5200045 |
| 6/7/2012 | ML12159A490 | GEH Application- ABWR Design Certification Renewal. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 7/9/2012 | ML12192A301 | Incoming YT-2012-0126 ABWR DC Renewal. | Concurrence on GEH Letter Non-Public | NRC/NRO | NRC/NRO | 5200045 |
| 7/20/2012 | ML12125A385 | GE Hitachi Nuclear Energy - U.S. Advanced Boiling Water Reactor Design Certification Renewal Application. | Project Plans and Schedules | NRC/NRO/DSRA/BPTS | NRC/NRO/DNRL/LB2 | 5200045 |
| 9/17/2012 | ML12261A311 | Response to NRC Letter: GE Hitachi Nuclear Energy – United States Advanced Boiling-Water Reactor Design Certification Renewal Application (July 20, 2012). | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/20/2012 | ML12321A262 | GEH Renewal Memo | E-Mail | NRC/NRO/DNRL/LB3 | | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 11/20/2012 | ML12321A205 | DSEA PRIMA FACIE Items for GEH ABWR Renewal | No Document Type Applies | NRC/NRO/DSEA/RPAC | | 5200045 |
| 11/20/2012 | ML12325A080 | ABWR GEH DCD Renewal Chapter 12 Table Errors Resolution Strategy. | No Document Type Applies | NRC/NRO/DSEA/RPAC | | 5200045 |
| 2/28/2013 | ML13059A302 | GE Hitachi Nuclear Energy Organization Chart. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/28/2013 | ML13059A303 | Enclsoure 1 - GEH Organization Chart. | NRC/Document Control Desk | | | 5200045 |
| 2/28/2013 | ML13059A304 | Enclosure 2 - GE-Hitachi Nuclear Energy Americas, LLC - Affidavit. | Organization Chart | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 10/31/2013 | ML13305B035 | NRC Review of GE Hitachi Nuclear Energy - US Advanced Boiling Water Reactor Design Certification Renewal Application. | Legal-Affidavit | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 3/17/2014 | ML14078A070 | GE Hitachi Nuclear Energy - United States Advanced Boiling Water Reactor (ABWR) Design Certification Renewal Application - Submittal Date for ABWR DCD Revision 6. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 3/17/2014 | ML14078A071 | ABWR Design Certification Renewal Application Revision 6 Schedule and Scope Description. | Attachment Issue Resolution Detail-Non-Public | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 4/24/2014 | ML14114A566 | Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 For GEH ABWR Design Certification Rule Renewal Application. | RAI Letter No. 1 | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 5/23/2014 | ML14147A422 | Initial Response to NRC's Request for Additional Information Letter Number 1 related to chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | Request for Additional Information (RAI) | NRC/NRO/DNRL/LB3 | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 6/26/2014 | ML14184B414 | Interim Part 21 Report - Containment Loads Potentially Exceed Limits with High Suppression Pool Water Level in the ABWR Design. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 7/16/2014 | ML14197A127 | Initial Response to NRC's Request for Additional Information Letter Number 1 related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/27/2014 | ML14239A139 | Enclosure 1 - MFN 14-052 - GEH Response to RAI 06.02.01.01.C-1 Proprietary Version. | Letter -Non-Public Proprietary | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/27/2014 | ML14239A140 | Enclosure 2 MFN 14-052 - GEH Response to RAI 06.02.01.01.C-1. | Letter | GE-Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 8/27/2014 | ML14239A138 | Enclosure 3 - MFN 14-052 - GEH Response to RAI 06.02.01.01.C-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/27/2014 | ML14239A141 | Enclosure 4 - MFN 14-052 - Affidavit. | Letter | GEH | NRC/NRO | 5200045 |

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| 8/27/2014 | ML14239A142 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 06.02.01.01.C-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/29/2014 | ML14241A558 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 08.02-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/29/2014 | ML14241A557 | Enclosure 2 to MFN 14-056, GEH Response to RAI 08.02-1, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 8/29/2014 | ML14241A559 | Enclosure 1 to MFN 14-056, GEH Response to RAI 08.02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 8/29/2014 | ML14241A306 | 10 CFR Part 21.21(a)(2) Closure Report Notification: Containment Loads Potentially Exceed Limits with High Suppression Pool Water Level in the ABWR Design. | No Document Type Applies | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 9/22/2014 | ML14118A096 | Letter - Request for Withholding Information from Public Disclosure (MFN 13-086. Supplement 1). | No Document Type Applies | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 9/24/2014 | ML14273A456 | Enclosure 3, GEH Response to RAIs 19-1, 19-2, 19-3, 19-4 and 19-5. | Letter | GEH | NRC/NRO | 5200045 |
| 9/24/2014 | ML14273A455 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, & 19 - GEH Responses to RAIs 19-1, 19-2, 19-3, 19-4 and 19-5. | Letter | GEH | NRC/NRO | 5200045 |

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| 9/25/2014 | ML14267A352 | Letter - Request for Additional Information Letter No. 2 Related to Chapter 1, 2 and 12 For GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 2 | NRC/NRO | GEH | 5200045 |
| 10/30/2014 | ML14303A405 | GEH RAI Schedule Dates in response to Request for Additional Information Letter Number 2 Related To Chapters 1, 2, And 12 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | Letter | GEH | NRC/NRO | 5200045 |
| 11/3/2014 | ML14309A024 | Enclosure 2: GEH Supplemental Response to RAI 19-5, ABWR DCD Revised Figures 9A.4-2, PA.4-5 and 9A.4-6. | Letter | GEH | NRC/NRO | 5200045 |
| 11/3/2014 | ML14309A023 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response to RAI 19-5. | Letter | GEH | NRC/NRO | 5200045 |
| 11/6/2014 | ML14310A568 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 01.05-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/6/2014 | ML14310A569 | Enclosure 2 to MFN 14-071 - GEH Response to RAI 01.05-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 11/6/2014 | ML14310A570 | Enclosure 1 to MFN 14-071 - GEH Response to RAI 01.05-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/19/2014 | ML14324A084 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/20/2014 | ML14324A083 | Enclosure 1 - MFN 14-075 - GEH Response to RAI 02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/20/2014 | ML14324A085 | Enclosure 2 - MFN 14-075 - GEH Response to RAI 02-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 12/16/2014 | ML14350A845 | Enclosure 2 - GEH Response to RAI 12.02-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 12/16/2014 | ML14350A844 | Enclosure 1 - GEH Response to RAI 12.02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 12/16/2014 | ML14350A846 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 12.02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 1/22/2015 | ML15023A018 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Responses to RAIs 12.02-2 and 12.02-3. | Letter | GEH | NRC/NRO | 5200045 |

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|---|---|---|---|---|---|---|
| 1/22/2015 | ML15023A020 | Enclosure 1 to MFN 15-003: GEH Responses to RAIs 12.02-2 and 12.02-3. | Letter | GEH | NRC/NRO | 5200045 |
| 1/22/2015 | ML15023A019 | Enclosure 2 to MFN 15-003: GEH Response to RAIs 12.02-2 and 12.02-3 - ABWR DCD Draft Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 1/23/2015 | ML15023A017 | Enclosure 3 to MFN 15-003: GEH Response to RAIS 12.02-2 and 12.02-3 - ABWR DCD Revised Figures. | Letter | GEH | NRC/NRO | 5200045 |
| 2/24/2015 | ML15056A146 | OEDO-15-00171 - Renee Taylor, OCM/JB Email re: Briefing Package for Meeting with General Electric on March 12, 2015. | E-Mail- Non-Public | NRC/OCM | NRC/NRO | 5200045 |
| 2/25/2015 | ML15056A129 | OEDO-15-00171-Ticket - Renee Taylor, OCM/JB Email re: Briefing Package for Meeting with General Electric on March 12, 2015. | E-Mail- Non-Public | NRC/OCM | NRC/EDO | 5200045 |
| 3/10/2015 | ML15068A227 | Request for Additional Information Letter No. 3 Related to Chapters 6 and 14 For GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 3 | NRC/NRO | GEH | 5200045 |
| 3/12/2015 | ML15069A674 | Request for Additional Information Letter No. 4 Related to Chapter 11 For GE Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 4 | NRC/NRO | GEH | 5200045 |
| 3/12/2015 | ML15062A427 | OEDO-15-00171-Ticket - Response Renee Taylor, OCM/JB Briefing Package for Meeting with General Electric on March 12, 2015. | Briefing Package- Non-Public | NRC/NRR | NRC/EDO | 5200045 |

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|---|---|---|---|---|---|---|
| 4/1/2015 | ML15093A532 | OEDO-15-00197 Response to Input for the INFO Digest - Appendix B New Reactor Application Chart (NRO). | Spreadsheet File-Non-Public | NRC/NRO | NRC/EDO/AO | 5200045 |
| 4/1/2015 | ML15092A177 | GEH Response to RAI 14.03-1, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 4/1/2015 | ML15092A176 | GEH Response to RAI 14.03-1. | Letter | GEH | NRC/NRO | 5200045 |
| 4/1/2015 | ML15092A178 | GE-Hitachi Nuclear Energy, Response to NRC Request for Additional Information RAI 14.03-1, Letter Number 3 Related to Chapters 6 and 14 Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Letter | GEH | NRC/NRO | 5200045 |
| 4/8/2015 | ML15098A485 | Enclosure 2 to MFN 15-024, GEH Response to RAI 06.03-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 4/8/2015 | ML15098A486 | Enclosure 1 to MFN 15-024, GEH Response to RAI 06.03-1. | Letter | GEH | NRC/NRO | 5200045 |
| 4/8/2015 | ML15098A487 | NRC Request for Additional Information Letter Number 3 Related to Chapters 6 and 14 for GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 06.03-1. | Letter | GEH | NRC/NRO | 5200045 |
| 4/9/2015 | ML15099A589 | GEH, Response to RAI 11.04-1 - ABWR DCD Draft Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |

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| 4/9/2015 | ML15099A590 | GEH, Response to RAI 11.04-1. | Letter | GEH | NRC/NRO | 5200045 |
| 4/9/2015 | ML15099A588 | GE-Hitachi Nuclear Energy, Response to RAI 11.04-1 Letter Number 4 Related to Chapter 11 Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | Letter | GEH | NRC/NRO | 5200045 |
| 4/20/2015 | ML15110A122 | Request for Additional Information Letter Number 5 Related to Chapters 6, 7, and 19 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 5 | GEH | NRC/NRO | 5200045 |
| 4/29/2015 | ML15118A725 | Request for Additional Information Letter No. 6 Related to Chapter 9 For GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 6 | NRC/NRO | GEH | 5200045 |
| 5/5/2015 | ML15125A258 | Transmittal of GEH Presentation Material for GEH/NRC May 7th, 2015 Advanced Boiling Water Reactor Design Certification Renewal Application Request for Additional Information Meeting. | Letter | GEH | NRC/NRO | 5200045 |
| 5/5/2015 | ML15125A259 | GEH Presentation Related to the GEH Responses to NRC Requests for Additional Information on the ABWR Design Certification Renewal Application for the May 7th Closed Session. | Letter-Non-Public Proprietary | GEH | NRC/NRO | 5200045 |
| 5/5/2015 | ML15125A260 | GEH Presentation Related to the GEH Responses to NRC Requests for Additional Information on the ABWR Design Certification Renewal Application for the May 7th Public Session. | Letter | GEH | NRC/NRO | 5200045 |

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|---|---|---|---|---|---|---|
| 5/5/2015 | ML15125A257 | Enclosure 3 - MFN 15-031 Affidavit. | Legal-Affidavit | GEH | NRC/NRO | 5200045 |
| 5/7/2015 | ML15162A635 | Presentation - Closed Meeting to Discuss Requests for Additional Information Associated with the Review of the ABWR Certification Renewal on May 7, 2015. | Letter -Non-Public Proprietary | GEH | NRC/NRO | 5200045 |
| 5/19/2015 | ML15139A213 | Enclosure 2 to MFN 15-037, GEH Response to RAI 07-1 ABWR DCD DRAFT Revision 6 Markup. | Letter | GEH | NRC/NRO | 5200045 |
| 5/19/2015 | ML15139A212 | Enclosure 1 to MFN 15-037, GEH Response to RAI 07-1. | Letter | GEH | NRC/NRO | 5200045 |
| 5/19/2015 | ML15139A211 | Initial Response to NRC's Request for Additional Information Letter Number 5 related to Chapters 6, 7, and 19 for Design Certification Rule Renewal Application. | Letter | GEH | NRC/NRO | 5200045 |
| 5/27/2015 | ML15147A596 | Enclosure 2 - GEH Response to Item #12 ABWR DCD DRAFT Revision 6 Markup. | Letter | GEH | NRC/NRO | 5200045 |
| 5/27/2015 | ML15147A595 | Enclosure 1 - GEH Response to Item #12. | Letter | GEH | NRC/NRO | 5200045 |
| 5/27/2015 | ML15147A594 | GEH Proposed Resolution of Item #12 of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |

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|---|---|---|---|---|---|---|
| 5/29/2015 | ML15149A234 | Enclosure 1 - GEH Response to RAI 06.02.01-1. | Letter | GEH | NRC/NRO | 5200045 |
| 5/29/2015 | ML15149A235 | Enclosure 2 - GEH Response to RAI 06.02.01-1 - ABWR DCD DRAFT Revision 6 Markup. | Letter | GEH | NRC/NRO | 5200045 |
| 5/29/2015 | ML15149A233 | Response to NRC's Request for Additional Information Letter Number 5 Related to Chapters 6, 7, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | Letter | GEH | NRC/NRO | 5200045 |
| 6/9/2015 | ML15154B692 | Request for Additional Information Letter Number 7 Related to Chapter 8 For GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 7 | NRC/NRO | GEH | 5200045 |
| 6/9/2015 | ML15160A421 | Request for Additional Information Related to Chapter 2 For GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 8 | NRC/NRO | GEH | 5200045 |
| 6/18/2015 | ML15170A045 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response to RAI 01.05-1. | Request for Additional Information (RAI) | NRC/NRO | GEH | 5200045 |
| 6/19/2015 | ML15170A041 | Enclosure 1: GE Hitachi Nuclear Energy - GEH Response to Item #3. | Letter | GEH | NRC/NRO | 5200045 |

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|---|---|---|---|---|---|---|
| 6/19/2015 | ML15170A036 | Enclosure 1 to MFN 15-044, GEH Response to Item #17. | Letter | GEH | NRC/NRO | 5200045 |
| 6/19/2015 | ML15170A035 | GEH Proposed Resolution of Item #17 of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 6/19/2015 | ML15170A047 | GEH Response and Supplemental Response to RAI 01.05-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 6/19/2015 | ML15170A046 | GEH Response and Supplemental Response to RAI 01.05-1. | Letter | GEH | NRC/NRO | 5200045 |
| 6/19/2015 | ML15170A037 | Enclosure 2 to MFN 15-044, GEH Response to Item #17, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 6/19/2015 | ML15170A042 | Enclosure 2: GE Hitachi Nuclear Energy - GEH Response to Item #3 - ABWR DCD Draft Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 6/19/2015 | ML15170A040 | GE Hitachi Nuclear Energy - GEH Proposed Resolution of Item #3 of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 6/26/2015 | ML15177A038 | GEH Response and Supplemental Response to RAI 02-1. | Letter | GEH | NRC/NRO | 5200045 |

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| 6/26/2015 | ML15177A044 | GEH Response to Item #13 - ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 6/26/2015 | ML15177A043 | GEH Response to Item #13 - Overhead Heavy Load Handling System Cranes. | Letter | GEH | NRC/NRO | 5200045 |
| 6/26/2015 | ML15177A039 | GEH Supplemental Response to RAI 02-1 - ABWR DCD Draft Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 6/26/2015 | ML15177A042 | GEH Proposed Resolution of Item #13 - Overhead Heavy Load Handling System Cranes of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 6/26/2015 | ML15177A037 | Supplemental Response to RAI 02-1 Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Letter | GEH | NRC/NRO | 5200045 |
| 7/6/2015 | ML15187A023 | GEH Proposed Resolution of Item # 4 - Maximum Groundwater Level of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 7/6/2015 | ML15187A025 | GEH Response to Item #4 - Maximum Groundwater Level ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/6/2015 | ML15187A024 | GEH Response to Item #4 - Maximum Groundwater Level. | Letter | GEH | NRC/NRO | 5200045 |

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| 7/7/2015 | ML15188A270 | GEH Proposed Resolution of Item # 28 - Fukushima Recommendation 9.3, Emergency Preparedness - of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A255 | GEH Proposed Resolution of Item # 10 - Gas Accumulation Locations -of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A272 | Enclosure 2, GEH Response to Item #28: Fukushima Recommendation 9.3, - Emergency Preparedness, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A263 | Enclosure 2 to MFN 14-063 Supplement 2: GEH Response and Supplemental Response #2 to RAI 19-5 ABWR DCD Draft Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A258 | Enclosure 2 - MFN 15-051 GEH Response to Item #10 - Gas Accumulation Locations ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A271 | Enclosure 1, GEH Response to Item #28: Fukushima Recommendation 9.3 - Emergency Preparedness. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A257 | Enclosure 1 - MFN 15-051 GEH Response to Item #10 - Gas Accumulation Locations. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A262 | Enclosure 1 to MFN 14-063 Supplement 2: GEH Response and Supplemental Response #2 to RAI 19-5. | Letter | GEH | NRC/NRO | 5200045 |

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| 7/7/2015 | ML15194A053 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Responses to RAIs 12.02-2 and 12.02-3. | Letter | GEH | NRC/NRO | 5200045 |
| 7/7/2015 | ML15188A261 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8 and 19 for GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response #2 to RAI 19-5. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A333 | NRC Request for Additional Information Letter Number 3 Related to Chapters 6 and 14 for GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response to RAI 06.03-1. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A102 | Request for Additional Information Letter Number 5 Related to Chapters 6, 7, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 19-6 through 19-9. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A103 | Enclosure 1, GEH Responses to RAIs 19-6 through 19-9. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A334 | Enclosure 1 to MFN 15-024, Supplement 1, GEH Response and Supplemental Response to RAI 06.03-1. | Letter | GEH | NRC/NRO | 5200045 |

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| 7/17/2015 | ML15198A104 | Enclosure 2, GEH Responses to RAIs 19-6 through 19-9, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A335 | Enclosure 2 to MFN 15-024, Supplement 1, GEH Supplemental Response to RAI 06.03.1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A345 | GEH Proposed Resolution of Item # 23 - Fuel Oil Transfer System of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A347 | GEH Response to Item #23 - Fuel Oil Transfer System, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/17/2015 | ML15198A346 | GEH Response to Item #23 - Fuel Oil Transfer System. | Letter | GEH | NRC/NRO | 5200045 |
| 7/20/2015 | ML15162A613 | Summary of Meeting To Discuss Requests For Additional Information Associated With The Review Of The ABWR Certification Renewal On May 7, | Meeting Summary | NRC/NRO | GEH | 5200045 |
| 7/21/2015 | ML15202A046 | RAI Letter Number 4 Related to Chapter 11 for GEH Supplemental Response to RAI 11.04-1. | Letter | GEH | NRC/NRO | 5200045 |
| 7/21/2015 | ML15202A047 | GEH Response and Supplemental Response to RAI 11.04-1. | Letter | GEH | NRC/NRO | 5200045 |

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| 7/21/2015 | ML15202A048 | GEH Supplemental Response to RAI 11.04-1 - ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 7/24/2015 | ML15209A561 | Request for Additional Information Letter Number 8 Related to Chapter 2 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 02.05.04-1. | Letter | GEH | NRC/NRO | 5200045 |
| 7/30/2015 | ML15212A762 | Request for Additional Information Letter Number 6 Related to Chapter 9 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 09.05.01-1. | Letter | GEH | NRC/NRO | 5200045 |
| 8/4/2015 | ML15216A312 | Transmittal of GEH Proposed Resolution of Item # 5 - Minimize Contamination - of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 8/4/2015 | ML15216A314 | Enclosure 2, GEH Response to Item #5 - Minimize Contamination, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 8/4/2015 | ML15216A313 | Enclosure 1, GEH Response to Item #5 - Minimize Contamination. | Letter | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B140 | Enclosure 1 to MFN 15-065, GEH Response to Items: #18b - Structural, Dynamic and Impact Analysis of New and Spent Fuel Racks, #19 - Thermal Hydraulic Analysis of the Spent Fuel Racks and #20 - Criticality Analyses of New and Spent Fuel Storage Racks. | Letter | GEH | NRC/NRO | 5200045 |

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| 8/11/2015 | ML15223B149 | Enclosure 2 to MFN 14-052, Revision 1, GEH Revised Response to RAI 06.02.01.01.C-1. | Letter | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B141 | Enclosure 2 to MFN 15-065, GEH ABWR DCD Draft Revision 6 Markups for Items 18b, 19 and 20. | Letter | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B150 | Enclosure 3 to MFN 14-052, Revision 1, GEH Revised Response to RAI 06.02.01.01.C-1 ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B148 | Enclosure 1 to MFN 14-052, Revision 1, GEH Revised Response to RAI 06.02.01.01.C-1. | Letter-Non-Public Proprietary | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B151 | Enclosure 4 to MFN 14-052, Revision 1, Affidavit. | Legal-Affidavit | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B139 | GEH Proposed Resolution of Items #18b, 19 and 20 of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 8/11/2015 | ML15223B147 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Revised Response to RAI 06.02.01.01.C-1. | Letter | GEH | NRC/NRO | 5200045 |

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|---|---|---|---|---|---|---|
| 8/14/2015 | ML15226A417 | Submittal of GEH Proposed Resolution to NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes, Item #21 - Obsolete Communication Technology. | Letter | GEH | NRC/NRO | 5200045 |
| 8/14/2015 | ML15226A419 | GEH Response to Item #21 - Obsolete Communication Technology. | Letter | GEH | NRC/NRO | 5200045 |
| 8/20/2015 | ML15230A204 | 8/13/15 - Summary of Meeting to Discuss Open Items Associated with The Review of the Advanced Boiling Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | GEH | 5200045 |
| 8/24/2015 | ML15236A229 | Enclosure 1, Revised GEH Response to Item #4, Maximum Groundwater Level. | Letter | GEH | NRC/NRO | 5200045 |
| 8/24/2015 | ML15236A227 | GEH Revised Proposed Resolution of Item # 4 - Maximum Groundwater Level of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 8/24/2015 | ML15236A230 | Enclosure 2, Revised GEH Response to Item #4, Maximum Groundwater Level, ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 8/25/2015 | ML15237A195 | GEH Response and Revised Supplemental Response to RAI 01.05-1 - ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |

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| 8/25/2015 | ML15237A194 | GEH Response and Revised Supplemental Response to RAI 01.05-1. | Letter | GEH | NRC/NRO | 5200045 |
| 8/25/2015 | ML15237A193 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Revision to Supplemental Response to RAI 01.05-1. | Letter | GEH | NRC/NRO | 5200045 |
| 9/9/2015 | ML15254A042 | GEH Proposed Resolution of Item # 26 - Fukushima Recommendation 4.2 Mitigation Strategies of NRC suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 9/11/2015 | ML15258A666 | GEH Proposed Resolution of Item #25 - Control Room Design with  State of the Art  HFE - NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 9/17/2015 | ML15264A003 | Request for Additional Information Letter Number 5 Related to Chapters 6, 7, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Revised Response to RAI 19-6 through 19-9. | Letter | GEH | NRC/NRO | 5200045 |
| 9/21/2015 | ML15267A060 | GE-Hitachi Nuclear Energy Americas, LLC - Proposed Resolution of Item No. 10 - Gas Accumulation Locations - of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes - Supplemental Response. | Letter | GEH | NRC/NRO | 5200045 |

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| 9/24/2015 | ML15271A169 | GE-Hitachi - Proposed Resolution of Item No. 18a - Seismic/Structural Analysis for Reactor Core of NRC Suggested U.S. Advance Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 9/25/2015 | ML15271A171 | GEH Proposed Resolution of Item Nos. 14, 15 and 16 - ABWR Probabilistic Risk Assessment - of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GEH | NRC/NRO | 5200045 |
| 9/25/2015 | ML15271A170 | NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 08.02-2. | Letter | GEH | NRC/NRO | 5200045 |
| 10/14/2015 | ML15274A398 | 10/28/15 - Audit Plan to Verify Evaluations Of Impact Of The Suppression Pool Hydrodynamic Loads. | Audit Plan | NRC/NRO | GEH | 5200045 |
| 10/29/2015 | ML15302A309 | Request for Additional Information Letter Number 6 Related to Chapter 9 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - Revision 1 to GEH Response to RAI 09.05.01-1. | Letter | GEH | NRC/NRO | 5200045 |
| 10/29/2015 | ML15302A311 | Revised GEH Response to RAI 09.05.01-1 - ABWR DCD DRAFT Revision 6 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 10/29/2015 | ML15302A310 | Revised GEH Response to RAI 09.05.01-1. | Letter | GEH | NRC/NRO | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 11/5/2015 | ML15309A158 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response #2 to RAI 02-01. | Letter | GEH | NRC/NRO | 5200045 |
| 11/5/2015 | ML15309A158 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2, and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response #2 to RAI 02-01. | Letter | GEH | NRC/NRO | 5200045 |
| 11/5/2015 | ML15309A159 | Enclosure 1 to MFN 14-075 Supplement 2: GEH Response and Supplemental Responses #1 and #2 to RAI 02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/5/2015 | ML15309A160 | Enclosure 2 to MFN 14-075 Supplement 2: GEH Supplemental Response #2 to RAI 02-1 and ABWR DCD Revision 5 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 11/9/2015 | ML15306A104 | 10/15/15 - Summary of Meeting To Discuss Open Items Associated With The Review Of The Advanced Boiling-Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | GE-Hitachi | 5200045 |
| 11/13/2015 | ML15317A095 | GEH, Supplemental Response to RAI 02.05.04-1, ABWR DCD Revision 5 Markup. | Letter | GEH | NRC/NRO | 5200045 |

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| 11/13/2015 | ML15317A094 | GEH, Supplemental Response to RAI 02.05.04-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/13/2015 | ML15317A093 | Request for Additional Information Letter Number 8 Related to Chapter 2 for GE Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response to RAI 02.05.04-1. | Letter | GEH | NRC/NRO | 5200045 |
| 11/19/2015 | ML15323A354 | Request for Additional Information Letter Number 5 to Related to Chapters 6, 7, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response #2 to RAI 19-9. | Letter | GEH | NRC/NRO | 5200045 |
| 11/19/2015 | ML15323A356 | Enclosure 2 to MFN 15-053, Supplement 2: GEH Supplemental Response #2 to RAI 19-9 - ABWR DCD Revision 5 Markups. | Letter | GEH | NRC/NRO | 5200045 |
| 11/19/2015 | ML15323A355 | Enclosure 1 to MFN 15-053, Supplement 2: GEH Response and Supplemental Response #1 and #2 to RAI 19-9. | Letter | GEH | NRC/NRO | 5200045 |
| 12/3/2015 | ML15337A119 | GE-Hitachi Nuclear Energy Americas, LLC - Submittal of 10 CFR § 50.46 Annual Report for the GE ABWR Standard Plant Design 2015. | Letter | GEH | NRC/NRO | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
|---|---|---|---|---|---|---|
| 12/4/2015 | ML16053A279 | ABWR SSAR Amendment 37, Revision 9, Part 1 of 6. | Annual Operating Report | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/4/2015 | ML16053A280 | ABWR SSAR Amendment 37, Revision 9, Part 2 of 6. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/4/2015 | ML16053A282 | ABWR SSAR Amendment 37, Revision 9, Part 3 of 6. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/4/2015 | ML16053A275 | ABWR SSAR Amendment 37, Revision 9, Part 4 of 6. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/4/2015 | ML16053A276 | ABWR SSAR Amendment 37, Revision 9, Part 5 of 6. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/4/2015 | ML16053A277 | ABWR SSAR Amendment 37, Revision 9, Part 6 of 6. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/4/2015 | ML16053A278 | ABWR Standard Safety Analysis Report, Volume 1 of 2. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 12/4/2015 | ML16053A283 | ABWR Standard Safety Analysis Report, Volume 2 of 2. | Final Safety Analysis Report (FSAR) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/4/2015 | ML16053A264 | GE Hitachi Nuclear Energy - Re-transmittal of Proprietary ABWR SSAR Amendment 37, Revision 9 in CD Format. | Legal-Affidavit | | | 5200045 |
| 12/15/2015 | ML15343A408 | Request for Additional Information Letter No. 9 Related to Chapter 6 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 9 | NRC/NRO | GEH | 5200045 |
| 1/8/2016 | ML16008A080 | Enclosure 1 to MFN 16-001: ABWR COPS Redesign - ABWR DCD Revision 5 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/8/2016 | ML16008A079 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Redesign of ABWR Containment Overpressure Protection System (COPS) Pipe Diameter. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/12/2016 | ML16012A292 | MFN 14-075 Supplement 3: GEH Supplemental Response #3 to RAI 02-1 - ABWR DCD Revision 5 Markups. | Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/12/2016 | ML16012A291 | GEH - Supplemental Response #3 to RAI 02 to Related to Ch. 1, 2, & 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/21/2016 | ML15357A292 | General Electric Hitachi Nuclear Energy U. S. Advanced Boiling Water Reactor Containment Hydrodynamic Loads Regulatory Audit Summary Report. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 1/22/2016 | ML16022A253 | GE-Hitachi Nuclear Energy - Enclosure 1, Supplemental Response to RAI 19-5, Description of Changes to Figures 9A.4-2 through 9A.4-6. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/22/2016 | ML16022A254 | GE-Hitachi Nuclear Energy - Enclosure 2, Supplemental Response to RAI 19-5, ABWR DCD Revised Figures 9A.4-2 thru 9A.4-6. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/22/2016 | ML16022A252 | GE-Hitachi Nuclear Energy - NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for the Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Supplemental Response No. 3 to RAI 19-5. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/28/2016 | ML16027A283 | 01/19/16 - Summary Of Teleconference To Discuss Open Items Associated With The Review Of The Advanced Boiling-Water Reactor Certification Renewal. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 2/18/2016 | ML16049A046 | GE-Hitachi Nuclear Energy - Enclosure 2, ABWR COPS Redesign - ABWR DCD Revision 5 Markups. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 2/18/2016 | ML16049A044 | GE-Hitachi Nuclear Energy - Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Redesign of ABWR Containment Overpressure Protection System (COPS) Pipe Diameter Revision 1. | Meeting Summary | NRC/NRO | | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 2/18/2016 | ML16049A045 | GE-Hitachi Nuclear Energy - Enclosure 1, Response to NRC's Request for Supplemental Information on ABWR COPS Redesign. | Memoranda | NRC/NRO/DNRL/LB3 | NRC/NRO/DNRL/LB3 | 5200045 |
| 2/19/2016 | ML16180A484 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 – Chapter 19 - Response to Severe Accident Policy Statement (Public). | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16180A485 | GE-Hitachi ABWR Design Control Document, Tier 1, Rev. 6 (Public). | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A086 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A089 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 01 - Introduction and General Description of Plant | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A090 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 02 - Site Characteristics | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A091 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 03 - Design of Structures, Components, Equipment, and Systems - Volume 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A092 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 03 - Design of Structures, Components, Equipment, and Systems - Volume 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 2/19/2016 | ML16181A258 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 03 - Design of Structures, Components, Equipment, and Systems - Volume 2. | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO/ DNRL/LB3 | 5200045 |
| 2/19/2016 | ML16081A093 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 04 - Reactor | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A094 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 05 - Reactor Coolant System and Connected Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A095 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 06 - Engineered Safety Features | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A096 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 07 - Instrumentation and Control Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A097 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 08 - Electric Power | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A098 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 09 - Auxiliary Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A099 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 10 - Steam and Power Conversion System | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 2/19/2016 | ML16081A100 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 11 - Radioactive Waste Management | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A101 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 12 - Radiation Protection | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A102 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 13 - Conduct of Operations | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A103 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 14 - Initial Test Program | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A104 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 15 - Accident and Analysis | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A105 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 16 - Technical Specifications | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A106 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 16 - Technical Specifications Bases | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A107 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 17 - Quality Assurance | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 2/19/2016 | ML16081A109 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 18 - Human Factors Engineering | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A110 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 19 - Response to Severe Accident Policy Statement | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A111 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 20 - Question and Response Guide | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A112 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16180A489 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 1 (Public). | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A113 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A114 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 3 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A116 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 2/19/2016 | ML16081A118 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 5 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A119 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 6 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16180A488 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 6 (Public). | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A120 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 7 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16180A487 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 7 (Public). | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A121 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 8 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16180A486 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Chapter 21 - Engineering Drawings - Volume 8 (Public). | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A087 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Tier 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 2/19/2016 | ML16081A088 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 6 - Tier 2 - Table of Contents | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 2/19/2016 | ML16081A268 | Submittal of ABWR Standard Plant Design Certification Renewal Application Design Control Document, Revision 6, Tiers 1 and 2. | License-Application for Design Certification | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 2/25/2016 | ML16050A486 | 2/5/16 - Summary of Teleconference to Discuss Open Items Associated with the Review Of The Advanced Boiling-Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | | 5200045 |
| 3/9/2016 | ML16069A022 | Initial Response to NRC's Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy ABWR Design Certification Rule Renewal Application. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO/ DNRL/LB3 | 5200045 |
| 3/16/2016 | ML16076A067 | GEH's Supplemental Response for Item # 5 - Minimize Contamination - of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Memoranda | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO/ DNRL/LB3 | 5200045 |
| 3/16/2016 | ML16076A068 | Enclosure 1 to MFN 15-063, Supplement 1: GEH's Supplemental Response to Item #5 - Minimize Contamination. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | | 5200045 |
| 3/16/2016 | ML16076A069 | Enclosure 2 to MFN 15-063, Supplement 1: GEH's Supplemental Response to Item #5 - Minimize Contamination - ABWR DCD Revision 6 Markups. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/11/2016 | ML16102A348 | Enclosure 2 to MFN 15-064, Revision 1, Supplement 1: GEH's Supplemental Response to RAI 09.05.01-1 - ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 4/11/2016 | ML16102A346 | Enclosure 1 to MFN 15-064, Revision 1, Supplement 1: GEH's Revised Response and Supplemental Information RAI 09.05.01-1. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 4/11/2016 | ML16102A345 | Transmittal of Response to RAI Letter Number 6 Related to Chapter 9 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - Supplement 1 of Revision 1 to GEH Response to RAI 09.05.01-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/19/2016 | ML16110A155 | GEH Response to NRC's Request for Supplemental Information on ABWR COPS Redesign. | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 4/19/2016 | ML16110A154 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Redesign of ABWR Containment Overpressure Protection System (COPS) Pipe Diameter - Revision 1, Supplement 1. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/20/2016 | ML16111A557 | Summary of Teleconference to Discuss Open Items Associated with the Review of the Advanced Boiling-Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | | 5200045 |
| 4/29/2016 | ML16120A042 | Supplemental Information for GEH's Response to Item #26 - Fukushima Recommendation 4.2 Mitigation Strategies of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 4/29/2016 | ML16120A043 | MFN 15-069, Supplement 1 - GEH Proposed Resolution and Response to NRC's Request for Supplemental Information to Item #26 - Fukushima Recommendation 4.2 - Mitigation Strategies. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 4/29/2016 | ML16120A044 | MFN 15-069, Supplement 1 - GEH's Response to NRC's Request for Supplemental Information to Item #26 - Fukushima Recommendation 4.2 - Mitigation Strategies ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 4/29/2016 | ML16120A033 | GE-Hitachi Nuclear Energy Americas, LLC - Response to NRC Post-Fukushima Recommendations. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 4/29/2016 | ML16120A032 | GE-Hitachi Nuclear Energy Americas, LLC - Supplemental Information for Response to Item No. 26 - Fukushima Recommendation 4.2 Mitigation Strategies of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/3/2016 | ML16124A049 | GE Hitachi Advanced Boiling Water Reactor DCD Public Meeting Summary. | Meeting Summary | NRC/NRO | | 5200045 |
| 5/6/2016 | ML16127A035 | Enclosure 2 MFN 14-052, Revision 1, Supplement 1, GEH Revised Response to RAI 06.02.01.01.C-1 ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 5/6/2016 | ML16127A034 | Enclosure 1 MFN 14-052, Revision 1, Supplement 1, GEH Supplemental Response to RAI 06.02.01.01.C-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/6/2016 | ML16127A033 | NRC Request for Additional Information Letter Number 1, Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response to RAI 06.02.01.01.C-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/24/2016 | ML16145A346 | Transmittal of NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application-GEH Response to RAI 08.02-2, Revision 1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/24/2016 | ML16144A784 | 05/19/2016 Summary of Teleconference with GE Hitachi Nuclear Energy on Open Items Associated with the Review of the Advanced Boiling Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | | 5200045 |
| 5/24/2016 | ML16145A347 | GE Hitachi Nuclear Energy Revised Response to RAI 08.02-2, Enclosure 1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/24/2016 | ML16145A348 | GE Hitachi Nuclear Energy, Revised Response to RAI 08.02-2. BWR DCD Revision 6 Markups, Enclosure 2. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 5/27/2016 | ML16148A104 | Enclosure 2 - MFN 16-034 GEHs Response to RAI 06.03-2 - ABWR DCD Revision 6 Markups. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/27/2016 | ML16148A103 | Enclosure 1 - MFN 16-034 GEHs Response to RAI 06.03-2. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/27/2016 | ML16148A102 | Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application GEH Response to RAI 06.03-2. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/31/2016 | ML16152A513 | Response to RAI 02.05.04-1 Letter Number 8 Related to Chapter 2 for GE Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - GEH Supplemental 2. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/31/2016 | ML16152A514 | GEH Response and Supplemental Response #2 To RAI 02.05.04-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 5/31/2016 | ML16152A515 | GEH Supplemental Response #2 to RAI 02.05.04-1 ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/3/2016 | ML16155A026 | Enclosure to MFN-16-035 - Table of New or Revised ITAAC in ABWR Design Certification Renewal Application. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 6/3/2016 | ML16155A025 | GE Hitachi Nuclear Energy - Transmittal of Item No. 24 on NRC List of Design Changes for Consideration in Advanced Boiling Water Reactor Design Certification Renewal. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/8/2016 | ML16160A067 | Request for Additional Information Letter Number 10 Related to Chapter 13 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 10 | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 6/16/2016 | ML16168A303 | GEH, Response to NRC?s Request for Supplemental Information # 2 on ABWR COPS Redesign. | Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/16/2016 | ML16168A304 | ABWR COPS Redesign - ABWR DCD Revision 6 Markups. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/16/2016 | ML16168A302 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Redesign of ABWR Containment Overpressure Protection System (COPS) Pipe Diameter - Revision 1, Supplement 2. | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/22/2016 | ML16174A180 | GEH Supplemental Response #2 to RAI 06.02.01.01.C-1 Letter Number 1, Related to Chapters 6, 8, and 19 for Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/22/2016 | ML16174A181 | GEH Supplemental Response #2 to RAI 06.02.01.01.C-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 6/22/2016 | ML16174A184 | MFN 14-052, Revision 1, Supplement 2, GHE Affidavit. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/22/2016 | ML16174A182 | GEH Revised Response #2 to RAI 06.02.01.01.C-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 6/22/2016 | ML16174A183 | GEH Revised Response #2 to RAI 06.02.01.01.C-1; ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 7/14/2016 | ML16194A147 | Inspection of the GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Aircraft Impact Assessment. | Letter | NRC/NRO/ DCIP/CEVB | GE-Hitachi Nuclear Energy Americas, LLC, | 5200045 |
| 7/21/2016 | ML16174A175 | Letter - Peak Cladding Temperature for GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Letter | NRC/NRO/ DNRL | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 8/19/2016 | ML19031C851 | MFN-16-059 - Peak Cladding Temperature of GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Letter | GE Hitachi Nuclear Energy | NRC/NRO/ DCIP | 5200045 |
| 8/22/2016 | ML16235A417 | Introduction to Revised Supplement to ABWR Design Certification Environmental Report. | No Document Type Applies | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/22/2016 | ML16224A286 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Aircraft Impact Assessment - Inspection Plan, 05200045/2016201. | Inspection Plan | NRC/NRO/ DNRL | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |

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| 8/22/2016 | ML16235A416 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Revised Supplement to ABWR Design Certification Environmental Report. | Environmental Report | | | 5200045 |
| 8/22/2016 | ML16235A418 | Applicant's Supplemental Environmental Report - Amendment to Standard Design Certification. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 8/24/2016 | ML16237A122 | GEH Response to NRC's Request for Supplemental Information #2 to Item #26 - Fukushima Recommendation 4.2 - Mitigation Strategies. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/24/2016 | ML16237A123 | GEH's Response to NRC's Request for Supplemental Information #2 to Item #26 - Fukushima Recommendation 4.2 - Mitigation Strategies. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/24/2016 | ML16237A121 | Supplemental Information #2 for GEH's Response to Item #26 - Fukushima Recommendation 4.2 Mitigation Strategies of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/30/2016 | ML16209A316 | GE Hitachi Nuclear Energy - United States Advanced Boiling Water Reactor Design Certification Renewal Review Schedule. | Letter | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 8/31/2016 | ML16244A125 | GE-Hitachi Nuclear Energy, Response to Request for Additional Information - ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 8/31/2016 | ML16244A123 | GE-Hitachi Nuclear Energy, Response to Request for Additional Information on Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Updated Review of Unresolved Safety Issues (USI), Generic Safety Issues (GSI), Generic... | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 8/31/2016 | ML16244A124 | GE-Hitachi Nuclear Energy, Response to Request for Additional Information on Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Updated Review of Unresolved Safety Issues (USI), Generic Safety Issues ... | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 9/2/2016 | ML16258A350 | GE-Hitachi nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - ABWR DCD Changes for Aircraft Impact Assessment (AIA) - Key Design Features. | Legal-Affidavit | | | 5200045 |
| 9/2/2016 | ML16258A351 | Enclosure 3: ABWR DCD Revision 6 Markups and ABWR DCD Revision 6 Ch. 21 Figure Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 9/14/2016 | ML16259A394 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - ABWR DCD Changes for Aircraft Impact Assessment (AIA) - Key Design Features (Revision 1). | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 9/22/2016 | ML16266A510 | 08/10/2016 Summary of Teleconference to Discuss Open Items Associated with the Review of the Advanced Boiling Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |

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| 9/30/2016 | ML16258A352 | Enclosure 4: Technical Report NEDE-33875P, ABWR US Certified Design, Aircraft Impact Assessment, and Licensing Basis Information and Design Details for Key Design Features. | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 9/30/2016 | ML16259A395 | Enclosure 2: Technical Report NEDE-33875P, Rev. 1, ABWR US Certified Design Aircraft Impact Assessment Licensing Basis Information and Design Details for Key Design Features. | Drawing | Hitachi-GE Nuclear Energy, Ltd | NRC/ Document Control Desk | 5200045 |
| 10/11/2016 | ML16285A133 | GEH, Response to NRC's Request for Supplemental Information #3 on ABWR COPS Redesign. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 10/11/2016 | ML16285A132 | GE-Hitachi Nuclear Energy, Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Redesign of ABWR Containment Overpressure Protection System (COPS) Pipe Diameter - Revision 1, Supplement 3. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/Document Control Desk | 5200045 |
| 10/12/2016 | ML16291A490 | Peak Cladding Temperature 2016 Annual Reporting Under 10 CFR § 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application. | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 10/25/2016 | ML16299A147 | GE-Hitachi Nuclear Energy - Supplement 1 to Proposed Resolution of Items #18b, 19 and 20 of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 10/25/2016 | ML16299A149 | MFN 15-065, Supplement 1 GEH ABWR DCD Draft Revision 7 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 10/25/2016 | ML16299A148 | MFN 15-065, Supplement 1 GEH Summary of Changes. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/15/2016 | ML16285A219 | The GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Aircraft Impact Assessment Inspection, Nuclear Regulatory Commission Inspection Report 05200045/2016201. | Inspection Report | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 11/16/2016 | ML16321A414 | GE Hitachi Nuclear Energy - Response to NRC Request for Additional Information Letter No. 2 Related to Chapters 1, 2, & 12 for Advanced Boiling Water Reactor Design Certification Rule Renewal Application Supplemental Response #4 to RAI 02-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/16/2016 | ML16321A415 | ABWR Design Certification Renewal Design Control Document Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/17/2016 | ML16323A005 | Enclosure 1: GEH's Response to NRC's Request for Supplemental Information covering the Updated Review of USIs, GSIs, GLs, NRC Bulletins and Operating Experience. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 11/17/2016 | ML16323A006 | Enclosure 2: MFN 16-065, Supplement 1 - GEH's Response to NRC's Request for Additional Information - ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 11/17/2016 | ML16323A004 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Updated Review of Unresolved Safety Issues (USI), Generic Safety Issues (GSI), Generic Letters (GL), NRC Bulletins and Operating Experience et al. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/21/2016 | ML16334A295 | NEDE-33875P, Revision 2, ABWR US Certified Design Aircraft Impact Assessment Licensing Basis Information and Design Details for Key Design Features. | Report, Technical | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/23/2016 | ML16334A294 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - ABWR DCD Changes for Aircraft Impact Assessment - Key Design Features (Revision 2): Enclosure 2, Revised ABWR DCD Rev 6 Chapter 21 Figure Markups. | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/23/2016 | ML16334A292 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - ABWR DCD Changes for Aircraft Impact Assessment - Key Design Features (Revision 2): Table of Figures & Revised ABWR DCD Revision 6 Markups. | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 11/30/2016 | ML16334A293 | NEDO-33875, Revision 0, ABWR US Certified Design Aircraft Impact Assessment Licensing Basis Information And Design Details For Key Design Features. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/6/2016 | ML16341A812 | GE Hitachi Advanced Boiling Water Reactor Design Certification Renewal Application, Actions Regarding Resolution of Item #26; Fukushima Recommendation 4.2, Mitigating Strategies. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/7/2016 | ML16342B005 | GE Hitachi Nuclear Energy - Reply to Notice of Violation, NRC Inspection Report 05200045/2016201. | Report, Technical | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/7/2016 | ML16342C334 | GE-Hitachi Nuclear Energy Americas, LLC - Response to 09.05.01-1 Request for Additional Information on Letter Number 6 Related to Chapter 9 Advanced Boiling-Water Reactor Design Certification Rule Renewal Application - Supplement 2 of Revision 1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/7/2016 | ML16342C338 | GE-Hitachi Nuclear Energy Americas, LLC - Supplemental Response to RAI 09.05.01-1 - ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/7/2016 | ML16342C337 | GE-Hitachi Nuclear Energy Americas, LLC - Revised Response and Supplemental Information to RAI 09.05.01-1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/13/2016 | ML16348A101 | GE-Hitachi Nuclear Energy Americas, LLC - Supplemental Information - ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/13/2016 | ML16348A099 | GE-Hitachi Nuclear Energy Americas, LLC - Supplemental Information for the Updated Review of USIs, GSIs, GLs, NRC Bulletins and Operating Experience. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 12/13/2016 | ML16348A097 | GE-Hitachi Nuclear Energy Americas, LLC - Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Updated Review of Unresolved Safety Issues, Generic Safety Issues, Generic Letters, NRC Bulletins & Operating Experience Supplement 2 | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/14/2016 | ML16349A174 | GEH - Revised Response to RAI 08.02-2 - ABWR DCD Revision 6 Markups. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/14/2016 | ML16349A173 | GEH - Revised Response to RAI 08.02-2. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/14/2016 | ML16349A172 | Response to NRC Request for Additional Information Letter Number 1 Related to Chapters 6, 8, and 19 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 08.02-02, Revision 2. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 12/19/2016 | ML16358A445 | Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application- GEH Response to RAI 06.03-2, Revision 1. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/11/2017 | ML17004A315 | 10/22/16 Summary of Teleconference with GE Hitachi Nuclear Energy to Discuss Open Items Associated With The Review Of The Advanced Boiling Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |

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| 1/11/2017 | ML17004A316 | 10/27/16 Summary of Teleconference with GE Hitachi Nuclear Energy to Discuss Open Items Associated with the Review of the Advanced Boiling Water Reactor Certification Renewal. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/23/2017 | ML17025A386 | GE Hitachi Nuclear Energy - Supplemental Information for Response to Item # 26 - Fukushima Recommendation 4.2 Mitigation Strategies -of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 1/25/2017 | ML17005A468 | GE Hitachi Nuclear Energy Response to Advanced Boiling Water Reactor Aircraft Impact Assessment Inspection, Nuclear Regulatory Commission Inspection Report No. 05200045/2016-201 and Notice of Violation. | Letter | NRC/NRO/DCIP/CEVB | GE Hitachi Nuclear Energy | 5200045 |
| 1/30/2017 | ML17031A058 | GE Hitachi Nuclear Energy - Transmittal of Revised Response, GEH Proposed Resolution of Item # 17 of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 1/30/2017 | ML17031A060 | Revised Response, GEH Proposed Resolution of Item #17, ABWR DCD Revision 6 Markups. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 1/30/2017 | ML17031A059 | Revised Response, GEH Proposed Resolution of Item #17. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/1/2017 | ML17033B598 | GE Hitachi Nuclear Energy - Reply to NRC Letter Regarding Inspection Report 05200045/2016-201 and Notice of Violation. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |

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| 2/2/2017 | ML17032A537 | Request for Additional Information Letter Number 1 Related to Environmental Assessment for Ge-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | Letter | NRC/NRR | GE Hitachi Nuclear Energy | 5200045 |
| 2/6/2017 | ML17037C992 | GE Hitachi Nuclear Energy Response to Advanced Boiling Water Reactor Aircraft Impact Assessment Inspection, Nuclear Regulatory Commission Inspection Report No. 05200045/2016-201 And Notice of Violation. | Request for Additional Information (RAI) | NRC/NRO/DNRL/LB3 | GE Hitachi Nuclear Energy | 5200045 |
| 2/15/2017 | ML17046A504 | U. S. Nuclear Regulatory Commission Regulatory Audit of ECCS Strainer Design As Part Of The ABWR Design Control Document. | Letter | NRC/NRO/ DCIP/CEVB | GE Hitachi Nuclear Energy | 5200045 |
| 2/16/2017 | ML17046A181 | GE-Hitachi Nuclear Energy - U.S. Advanced Boiling Water Reactor Design Certification Renewal Electronic Reading Room. | Audit Plan | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 2/23/2017 | ML17055C497 | Enclosure 5 - M170046 Technical Report NED0-33878 Public Information. | Letter | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 2/23/2017 | ML17055C500 | Enclosure 6 - M170046 Technical Report NEDE-33878P GEH Proprietary Information. | Report, Technical | NRC/NRO | GE Hitachi Nuclear Energy | 5200045 |
| 2/23/2017 | ML17055C495 | Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application- GEH Response to RAI 06.03-2, Revision 2. | Report, Technical | NRC/NRO | GE Hitachi Nuclear Energy | 5200045 |

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| 2/2/2017 | ML17032A537 | Request for Additional Information Letter Number 1 Related to Environmental Assessment for Ge-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | RAI Letter 1 | NRC/NRO | GEH | 5200045 |
| 2/6/2017 | ML17037C992 | GE Hitachi Nuclear Energy Response to Advanced Boiling Water Reactor Aircraft Impact Assessment Inspection, Nuclear Regulatory Commission Inspection Report No. 05200045/2016-201 And Notice of Violation. | Letter | NRC/NRO/ DCIP/CEVB | GE Hitachi Nuclear Energy | 5200045 |
| 2/15/2017 | ML17046A504 | U. S. Nuclear Regulatory Commission Regulatory Audit of ECCS Strainer Design as Part Of The ABWR Design Control Document. | Audit Plan | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 2/16/2017 | ML17046A181 | GE-Hitachi Nuclear Energy - U.S. Advanced Boiling Water Reactor Design Certification Renewal Electronic Reading Room. | Letter | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 2/23/2017 | ML17055C497 | Enclosure 5 - M170046 Technical Report NED0-33878 Public Information. | Report, Technical | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/23/2017 | ML17055C500 | Enclosure 6 - M170046 Technical Report NEDE-33878P GEH Proprietary Information. | Report, Technical | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |

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| 2/23/2017 | ML17055C495 | Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application- GEH Response to RAI 06.03-2, Revision 2. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/28/2017 | ML17059C519 | Response to NRC Feedback. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/28/2017 | ML17059C522 | Revised ABWR DCD Revision 6/Draft Revision 7 Markups. | Report, Miscellaneous | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/28/2017 | ML17059C525 | Technical Report NEDE-33875P, Revision 3. | Report, Technical | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/28/2017 | ML17059C523 | Technical Report NEDO-33875, Revision 3 Public. | Report, Technical | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 2/28/2017 | ML17059C517 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - ABWR DCD Changes for Aircraft Impact Assessment (AIA) - Key Design Features (Revision 3). | Letter | GE Hitachi Nuclear Energy | NRC/ Document Control Desk | 5200045 |
| 3/2/2017 | ML17061A068 | GE-Hitachi Nuclear Energy Americas, LLC, Revised Response to RAI 14.03-1. | Letter | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |

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| 3/2/2017 | ML17061A066 | GE-Hitachi Nuclear Energy Americas, LLC - Response to RAI 14.03-1, Rev. 1, Letter Number 3 Related to Chapters 6 and 14 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 3/2/2017 | ML17061A069 | GE-Hitachi Nuclear Energy Americas, LLC - Revised Response to RAI 14.03-1 ABWR DCD Markups. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 3/2/2017 | ML17060A476 | 03/02/2017 Presentation Slides for Public Meeting to Discuss ABWR Design Certification Renewal Application. | Meeting Briefing Package/Handouts | | | 5200045 |
| 3/20/2017 | ML17079A357 | GE Hitachi Nuclear Energy - ABWR DCD Markups (Supplemental Information on Peak Cladding Temperature/ 10 CFR § 50.46 for the Advanced Boiling Water Reactor Design Certification Renewal Application). | E-Mail | NRC/NRO | NRC/NRO | 5200045 |
| 3/20/2017 | ML17079A356 | GE Hitachi Nuclear Energy - Supplemental Information on Peak Cladding Temperature/ 10 CFR § 50.46 for the Advanced Boiling Water Reactor Design Certification Renewal Application. | Design Control Document (DCD) | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 3/20/2017 | ML17079A353 | GE Hitachi Nuclear Energy - Submittal of Supplemental Information on Peak Cladding Temperature/ 10 CFR § 50.46 for the Advanced Boiling Water Reactor | No Document Type Applies | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |

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| 3/21/2017 | ML17080A046 | GE-Hitachi Nuclear Energy Americas, LLC - Revised Response to RAI 14.03-1, ABWR DCD Markups. | Letter | GE Hitachi Nuclear Energy | NRC/ | 5200045 |
| 3/21/2017 | ML17080A045 | GE-Hitachi Nuclear Energy Americas, LLC - Revised Response to RAI 14.03-1. | Response to Request for Additional | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 3/21/2017 | ML17080A065 | GE-Hitachi Nuclear Energy - Response to NRC Environmental Assessment RAIs 01.05-2, 01.05-3, and 01.05-4. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 3/21/2017 | ML17080A064 | GE-Hitachi Nuclear Energy - Response to NRC Request for Additional Information 01.05-2, 01.05-3, and 01.05-4 Letter Number 1 Related to Environmental Assessment for Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/Document Control Desk | 5200045 |
| 3/21/2017 | ML17080A043 | GE-Hitachi Nuclear Energy Americas, LLC - Response to NRC Request for Additional Information 14.03-1 (Revision 2) Letter Number 3 Related to Chapters 6 and 14 for Advanced Boiling Water Reactor Design Certification Rule Renewal Application. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |
| 3/22/2017 | ML17080A456 | Notice of Planned Inspection of the GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Aircraft Impact Assessment. | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/Document Control Desk | 5200045 |
| 3/23/2017 | ML17082A215 | Summary of Teleconference to Discuss Open Items Associated with the Review of the Advanced Boiling-Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | | 5200045 |

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| 3/28/2017 | ML17087A290 | Request for Additional Information Letter No. 11 - GEH Advanced Boiling Water Reactor DC Renewal. | RAI Letter No. 11 | NRC/NRO | GEH | 5200045 |
| 4/7/2017 | ML17095A848 | Inspection Plan for the GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Aircraft Impact Assessment. | Inspection Plan | | | 5200045 |
| 4/7/2017 | ML17090A308 | Memo Safety Evaluation With No Open Items GEH Advanced Boiling Water Reactor DC Renewal Section 12.3 Radiation Protection Design Features. | Memoranda | NRC/NRO/ DCIP/QVIB1 | NRC/NRO/ DCIP/QVIB1 | 5200045 |
| 4/7/2017 | ML17090A482 | Memo - Advanced Boiling Water Reactor Design Certification Renewal Application Safety Evaluation with No Open Items for Section 2.6.8, Requirements For Determination of ABWR Site Acceptability. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 4/11/2017 | ML17090A483 | Letter Safety Evaluation with No Open Items GEH ABWR DC Renewal Section 2.6.8. | Memoranda | NRC/NRO/ | NRC/ACRS | 5200045 |
| 4/11/2017 | ML17090A309 | Letter - Safety Evaluation With No Open Items GEH ABWR DC Renewal Section 12.3 Radiation Protection Design | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 4/13/2017 | ML17103A125 | Enclosure 1 to M170090 - GEH Supplemental Response # 5 to RAI 02-1. | Letter | GEH | NRC/NRO | 5200045 |

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| 4/13/2017 | ML17103A124 | NRC Request for Additional Information Letter Number 2 Related to Chapters 1, 2 and 12 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Supplemental Response #5 to RAI 02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 4/25/2017 | ML17116A071 | Forwards Response to NRC Request for Additional Information Letter Number 11 Re Chapter 6 for GE Hitachi Nuclear Energy, Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 06.03-3. | Letter | GEH | NRC/NRO | 5200045 |
| 4/25/2017 | ML17144A249 | Letter Safety Evaluation With No Open Items GEH ABWR DC Renewal Section 11.4 52417. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 5/3/2017 | ML17125A041 | Request Approval for Reassignment of Reviewing Officials. | NRC/NRO | | | 5200045 |
| 5/10/2017 | ML17130A798 | Request for Additional Information Letter Number 12 Related to Chapter 6 For GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application. | RAI Letter No. 12 | NRC/NRO/ DNRL | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 5/11/2017 | ML17132A027 | Request for Additional Information Letter Number 8 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling-Water Reactor Design Certification Rule Renewal Application-Review 1 of Report NED-33878 (Relates to GEH Response to RAI 06.03-2). | Letter | GE-Hitachi Nuclear Energy Americas, LLC | NRC/Document Control Desk | 5200045 |

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| 5/25/2017 | ML17144A278 | Letter Safety Evaluation with No Open Items GEH ABWR DC Renewal Section 2.5. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 5/25/2017 | ML17144A279 | Memo Safety Evaluation with No Open Items GEH Advanced Boiling Water Reactor DC Renewal Section 2.5 52417. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 5/25/2017 | ML17144A250 | Memo Safety Evaluation with No Open Items GEH Advanced Boiling Water Reactor DC Renewal Section 11.4 Solid Waste Management System 52417. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 5/31/2017 | ML17132A030 | Enclosure 2 to M170126 - Technical Report NEDE-33878P, Revision 1, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability . | Report, Technical | GE-Hitachi Nuclear Energy Americas, LLC | | 5200045 |
| 5/31/2017 | ML17132A028 | Enclosure 1 to M170126 - Technical Report NEDO-33878, Revision 1, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, Public Information. | Report, Technical | GE-Hitachi Nuclear Energy Americas, LLC | | 5200045 |
| 6/2/2017 | ML17150A478 | Memo Safety Evaluation With No Open Items GEH Advanced Boiling Water Reactor DC Renewal Section 19.2.3.3.4 ABWR Containment Vent Design . | Memoranda | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/2/2017 | ML17150A479 | Letter Safety Evaluation With No Open Items GEH ABWR DC Renewal Subection 19.2.3.3.4. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 6/16/2017 | ML17167A161 | NRC Request for Additional Information Letter Number 12 Related to Chapter 6 for GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Response to RAI 06.02.02-1. | Letter | GEH | NRC/NRO | 5200045 |
| 7/10/2017 | ML17187A127 | Request for Additional Information Letter No. 13 - ABWR. | RAI Letter No. 13 | NRC/NRO | GEH | 5200045 |
| 7/13/2017 | ML17193A396 | The GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Aircraft Impact Assessment Inspection Follow-Up, Nuclear Regulatory Commission Inspection Report No. 05200045/2017-201. | Inspection Report | NRC | GEH | 5200045 |
| 7/25/2017 | ML17205A054 | The United States Nuclear Regulatory Commission Plan for The Audit Of The Advanced Boiling Water Reactor Design Certification Renewal Application Probabilistic Risk Assessment Impact From Proposed Design Changes. | Letter | NRC/NRO | GEH | 5200045 |
| 7/25/2017 | ML17205A055 | Audit Plan - Advanced Boiling Water Reactor Design Certification Renewal Application, Probabilistic Risk Assessment Impact from Proposed Design Changes. | Request for Additional Information (RAI) | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 8/3/2017 | ML17199G648 | Letter - GE Hitachi Nuclear Energy - United States Advanced Boiling Water Reactor Design Certification Renewal Review Schedule. | Inspection Report | NRC | GEH | 5200045 |

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| 8/23/2017 | ML17236A061 | Enclosure 1 - GEH Revised Response to RAI 06.03-3 and Responses to RAIs 06.03-4 through 06.03-9. | Letter | NRC/NRO/ DCIP/QVIB1 | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |
| 8/23/2017 | ML17236A062 | Enclosure 2 - ABWR DCD Markups for RAI Responses. | Letter | NRC/NRO/ DNRL/LB3 | GE Hitachi Nuclear Energy | 5200045 |
| 8/23/2017 | ML17236A060 | NRC Requests for Additional Information Letter Nos. 11 and 13 Related to Chapter 6 for GE Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application - GEH Revised Response to RAI 06.03-3 and Responses to RAIs... | Letter | | | 5200045 |
| 8/31/2017 | ML17236A063 | Enclosure 3 - NEDO-33878, Revision 2, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, August 2017. | Response to Request for Additional Information (RAI) | GE Hitachi Nuclear Energy | NRC/ Document Control Desk | 5200045 |
| 8/31/2017 | ML17236A064 | Enclosure 4 - NEDE-33878P, Revision 2, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability, August 2017. | Letter | GEH | NRC/NRO | 5200045 |
| 10/10/2017 | ML17283A305 | GE Hitachi Nuclear Energy - ABWR DCD Markups for COL Information Items for Mitigating Strategies. | Response to Request for Additional Information (RAI) | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |
| 10/10/2017 | ML17283A307 | GE Hitachi Nuclear Energy - ABWR Design Certification Annual 10 CFR § 50.46 Report for 2017. | Response to Request for Additional Information (RAI) | GE Hitachi Nuclear Energy | NRC/NRO | 5200045 |

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|---|---|---|---|---|---|---|
| 10/24/2017 | ML17297B240 | GE-Hitachi Nuclear Energy Advanced Boiling Water Reactor Design Certification Rule Renewal Application-GEH Request for Exemption. | Design Control Document (DCD) | GEH | NRC/NRO | 5200045 |
| 11/9/2017 | ML17311A055 | Summary of Public Meeting Teleconference with GEH Regarding the Advanced Boiling Water Renewal. | Meeting Summary | NRC/NRO | GEH | 5200045 |
| 12/4/2017 | ML17320A188 | Memo Safety Evaluation with No Open Items GEH Advanced Boiling Water Reactor DC Renewal Subsections 14.3.2.3.6 and 14.3.2.3.8. | Memoranda | NRC/NRO/ DNRL/LB3 | NRC/NRO/ DNRL/LB3 | 5200045 |
| 12/4/2017 | ML17338A133 | GE Hitachi Nuclear Energy - ABWR DCD Markups for AC-Independent Water Addition System Changes. | Memoranda | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 12/6/2017 | ML17320A190 | Letter Safety Evaluation with No Open Items GEH ABWR DC Renewal Subsections 14.3.2.3.6 and 14.3.2.3.8. | NRC/NRO | | | 5200045 |
| 12/11/2017 | ML17338A085 | Letter Safety Evaluation with No Open Items GEH ABWR DC Renewal Subsection 2.6.2 (002). | E-Mail | NRC/NRO/ DNRL | NRC/NRO | 5200045 |
| 12/11/2017 | ML17338A086 | Memo to ACRS - GEH ABWR DC Renewal SE for Subsection 2.6.2. | Letter | NRC/NRO/ DNRL/LB3 | GE-Hitachi Nuclear Energy Americas, | 5200045 |

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| 12/11/2017 | ML17080A134 | Chapter 2.0 - Site Characteristics, Section 2.6.2, Safety Evaluation With No Open Items, GEH Advanced Boiling Water Reactor DC Renewal. | Safety Evaluation Report | NRC/NRO/DNRL | | 5200045 |
| 1/16/2018 | ML17352A576 | Regulatory Audit Results Summary Report of The Probabilistic Risk Assessment Of Design Changes For The Advanced Boiling-Water Reactor Design Certification Renewal | Audit Report | | | 5200045 |
| 1/31/2018 | ML18029A259 | Memo to ACRS - GEH ABWR DC Renewal SE for Section 12. | Meeting Summary | NRC/NRO/DNRL | NRC/NRO/DNRL | 5200045 |
| 2/1/2018 | ML18029A260 | Letter to Applicant - GEH ABWR DC Renewal SE for Section 12. | Memoranda | NRC/NRO/DNRL | NRC/ACRS | 5200045 |
| 2/1/2018 | ML17065A197 | Chapter 12, Radiation Protection, Section 12.2, Safety Evaluation with No Open Items, GEH Advanced Boiling Water Reactor DC Renewal. | Letter | NRC/NRO/DNRL/LB3 | GE Hitachi Nuclear Energy | 5200045 |
| 2/2/2018 | ML17097A470 | GE Hitachi Nuclear Energy - U.S. Advanced Boiling-Water Reactor Design Certification Renewal Application, Closure Of Design Items 14, 15, 16, 21, 24 And 25 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/DNRL | NRC/ACRS | 5200045 |
| 3/2/2018 | ML18061A090 | GE Hitachi Nuclear Energy - Request Approval for Reassignment of Reviewing Officials. | Letter | NRC/NRO | GE-Hitachi Nuclear Energy Americas, LLC | 5200045 |

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| 3/6/2018 | ML18059A344 | Advanced Boiling-Water Reactor Design Certification Renewal Application Safety Evaluation with No Open items for Section 9.5.1, Fire Protection . | Memoranda | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 3/8/2018 | ML17354A814 | Chapter 9.5.1, Fire Protection System, Safety Evaluation with No Open Items, GEH Advanced Boiling Water Reactor DC Renewal. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 3/8/2018 | ML18059A342 | Letter to Applicant - GEH ABWR DC Renewal SE for Section 9.5.1 Final. | Letter | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 3/28/2018 | ML18092A303 | GE-Hitachi Nuclear Energy Americas, LLC - Advanced Boiling Water Reactor Design Certification Rule Renewal Application - Additional Information for Response to RAIs 06.03-4 through 06.03-9. | Legal-Affidavit | GEH | NRC/NRO | 5200045 |
| 3/31/2018 | ML18092A306 | NEDO-33878, Revision 3, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability. | Letter Topical | GEH | NRC/NRO | 5200045 |
| 3/31/2018 | ML18092A308 | NEDE-33878P, Revision 3, ABWR ECCS Suction Strainer Evaluation of Long-Term Recirculation Capability. | Response to Request for Additional Information (RAI) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 4/9/2018 | ML18088A224 | Memo - Safety Evaluation with No Open Items GEH Advanced Boiling Water Reactor Section 13.3, Emergency Planning . | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 4/10/2018 | ML18088A222 | 04/10/2018 Letter Re: Safety Evaluation with No Open Items for Section 13.3, Emergency Planning. | Report, Technical | GE-Hitachi Nuclear Energy Americas, LLC | NRC/NRO | 5200045 |

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| 5/1/2018 | ML18114A035 | Memo - Advanced Boiling-Water Reactor Design Certification Renewal Application Safety Evaluation with No Open Items for Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection | Memoranda | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 5/2/2018 | ML18114A036 | Letter - Advanced boiling-Water Reactor Design Certification Renewal Application Safety Evaluation With No Open Items For Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection | Letter | NRC/NRO/ DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 5/7/2018 | ML18052A137 | ABWR DC Renewal SER Section 5.2.5 Low Level Reactor Coolant Leakage Rev 2. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 6/11/2018 | ML18157A215 | 03/01/2018 Summary of Teleconference To Discuss Open Items Associated With The Review Of The Advanced Boiling Water Reactor Certification Renewal. | Meeting Summary | NRC/NRO | GEH | 5200045 |
| 6/12/2018 | ML18155A213 | Memo - GEH ABWR DC Renewal SE for Section 2.3 Meteorology. | E-Mail | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 6/12/2018 | ML18155A236 | Memo - GEH ABWR DC Renewal SE for Section 3.3 Severe Wind and Extreme Wind (Tornado and Hurricane) Loadings. | Memoranda | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |

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| 6/12/2018 | ML18155A292 | Memo - GEH ABWR DC Renewal SE for Section 3.5.1.4 Missiles Generated by Natural Phenomena | Memoranda | NRC/NRO | NRC/ACRS | 5200045 |
| 6/13/2018 | ML18155A212 | Letter to applicant - GEH ABWR DC Renewal SE for Section 2.3 Meteorology. | Letter | NRC/NRO/DNRL | GE Hitachi Nuclear Energy | 5200045 |
| 6/13/2018 | ML18155A235 | Letter to Applicant - GEH ABWR DC Renewal SE for Section 3.3 Severe Wind and Extreme Wind (Tornado and Hurricane) Loadings. | Letter | NRC/NRO/DNRL | NRC/ACRS | 5200045 |
| 6/13/2018 | ML18155A290 | Letter to applicant - GEH ABWR DC Renewal SE for Section 3.5.1.4 Missiles Generated by Natural Phenomena. | Letter | NRC/NRO | NRC/ACRS | 5200045 |
| 6/20/2018 | ML18026A750 | Advanced Boiling Water Reactor DC Renewal Safety Evaluation Section 2.3 - Meteorology. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/DNRL | NRC/ACRS | 5200045 |
| 6/20/2018 | ML18026A667 | Advanced Boiling Water Reactor Safety Evaluation, Section 3.3 - Tornado and Hurricane Loadings. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/DNRL | NRC/ACRS | 5200045 |
| 6/20/2018 | ML18026A776 | ABWR DC Renewal SER Section 3.5.1.4. | Safety Evaluation | NRC/NRO/DNRL | NRC/ACRS | 5200045 |

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| 6/21/2018 | ML18057A480 | ABWR DC Renewal SE Section 13.3 - Emergency Planning. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 6/22/2018 | ML18173A050 | GEH Proposed Resolution of Item #10 - Gas Accumulation Locations - of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes - Supplemental Response. | Letter | GE Hitachi Nuclear Energy | NRC/ Document Control Desk | 5200045 |
| 6/27/2018 | ML18176A143 | Memo to ACRS - GEH ABWR DC Renewal SE for Section 13.5. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/28/2018 | ML18176A144 | Letter to Applicant - GEH ABWR DC Renewal SE for Section 13. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/2/2018 | ML18046A992 | Chapter 13, Plant Procedures, Section 13.5 Safety Evaluation With No Open Items, GEH Advanced Boiling Water Reactor DC Renewal. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/11/2018 | ML18183A433 | Advanced Boiling-Water Reactor Design Certification Renewal Application Safety Evaluation With No Open Items For Section 9.1.4, Light Load Handling | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/11/2018 | ML18183A372 | Memo - GEH ABWR DC Renewal SE for Section 9.1.1. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/11/2018 | ML18183A418 | Memo - GEH ABWR DC Renewal SE for Section 9.1.5. | Memoranda | NRC/NRO/ DLSE | NRC/ACRS | 5200045 |

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| 7/12/2018 | ML18183A371 | Letter to applicant - GEH ABWR DC Renewal SE for Section 9.1.1. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/12/2018 | ML18183A417 | Letter to applicant - GEH ABWR DC Renewal SE for Section 9.1.5. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/12/2018 | ML18183A432 | Advanced Boiling-Water Reactor Design Certification Renewal Application - Safety Evaluation With No Open Items For Section 9.1.4, Light Load Handling System . | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/23/2018 | ML18096A046 | ABWR DC Renewal SER Section 9.1.1 - New Fuel Storage. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/23/2018 | ML18096A120 | ABWR DC Renewal SER Section 9.1.4 - Light Load Handling. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/23/2018 | ML18096A059 | ABWR DC Renewal SER Section 9.1.5 - Heavy Load Cranes. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 7/25/2018 | ML18199A273 | GEH ABWR DC Renewal Audit Plan - PCT Increase. | Audit Plan | NRC/NRO | GEH | 5200045 |
| 7/26/2018 | ML18029A130 | Chapter 3, Seismic Subsystem Analysis, Section 3.7.3, GEH Advanced Boiling Water Reactor DC Renewal. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/26/2018 | ML18200A319 | Letter to Applicant - GEH ABWR DC Renewal SE for Subsection 3.7.3. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 7/26/2018 | ML18200A320 | Memo to ACRS - GEH ABWR DC Renewal SE for Subsection 3.7.3. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 10/29/2018 | ML18302A023 | GE-Hitachi Nuclear Energy Americas, LLC - ABWR Design Certification Annual 10 CFR § 50.46 Report for 2018. | Memoranda | GEH | NRC/NRO | 5200045 |
| 11/1/2018 | ML18290A942 | OEDO-18-00517- Briefing Package for Meeting with GE Hitachi on November 1, 2018. | Annual Operating Report Non-Public | NRC/NRO | NRC/EDO | 5200045 |
| 11/13/2018 | ML18275A179 | Memo - GEH ABWR DC Renewal Application Safety Evaluation with no Open Items for Section 19.5, Aircraft Impact Assessment. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 11/15/2018 | ML18275A177 | Letter to Applicant - GEH ABWR DC Renewal SE for Section 19.5. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 11/15/2018 | ML18275A351 | ABWR DC Renewal SER Sect 19G-5 AIA Evaluation. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 12/4/2018 | ML18312A163 | Letter to Applicant - ABWR DC Renewal Safety Evaluation, Section 19.1, Probabilistic Risk Assessment. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 12/4/2018 | ML18312A164 | Memo - GEH ABWR DC Renewal Safety Evaluation for Section 19.1, Probabilistic Risk Assessment. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 12/4/2018 | ML18312A162 | Safety Evaluation - ABWR DC Renewal Safety Evaluation, Section 19.1, Probabilistic Risk Assessment. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 12/14/2018 | ML18346A554 | Memo - GEH ABWR DC Renewal Issue 10 SE for Chapter 5, Reactor Coolant System and Connected Systems, Section 5.4.8. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 12/19/2018 | ML18346A561 | Letter to applicant - GEH ABWR DC Renewal Issue 10 SE for Section 5.4.8. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 12/19/2018 | ML18352A788 | December 19, 2018 Presentation Slides - ABWR Certification Renewal Peak Cladding Temperature Increase NRC Public Meeting. | Slides and Viewgraphs | GE Hitachi Nuclear Energy | NRC/NRO/ DLSE | 5200045 |
| 1/11/2019 | ML19002A393 | Memo to ACRS - GEH Advanced Boiling Water Reactor DC Renewal Safety Evaluation for Subsections 8.2.5 and 8.3.3.17. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 1/15/2019 | ML19002A391 | Letter To Applicant - GEH ABWR DC Renewal SE for Subsections 8.2.5 and 8.3.3.17. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 1/15/2019 | ML18346A609 | ABWR DC Renewal SE Section 5.4.8 Reactor Water Cleanup System. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 1/21/2019 | ML19021A015 | GE-Hitachi Nuclear Energy Americas, LLC - Peak Cladding Temperature/10 CFR § 50.46 for Advanced Boiling Water Reactor (ABWR) Design Certification Renewal Application, Supplemental Information. | Letter | GEH | NRC/NRO | 5200045 |
| 1/22/2019 | ML19009A413 | Memo - Advanced Boiling Water Reactor Certification Renewal Peak Cladding Temperature Increase Public Meeting on December 19, 2018. | Meeting Summary | NRC/NRO | GEH | 5200045 |
| 1/24/2019 | ML18354B167 | ABWR DC Renewal - ECCS Suction Strainer Design Audit Report. | Audit Report | NRC/NRO | GEH | 5200045 |
| 2/4/2019 | ML18163A244 | GE Hitachi Advanced Boiling Water Reactor DC Renewal Proprietary Review Letter Response to Requests for Additional Information SSP Long Term Recirculation Capability. | Letter | GEH | NRC/NRO | 5200045 |
| 2/4/2019 | ML18234A432 | Memo SE With No Open Items GEH ABWR Subsection 6.2.1.6. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 2/4/2019 | ML18164A110 | Memo - GEH ABWR DC Renewal SE for Subsection 6.2.1.3. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 2/4/2019 | ML18234A431 | Letter Safety Evaluation with No Open Items GEH ABWR DC Renewal Subsection 6.2.1.6. | Letter | NRC/NRO/ DLSE | NRC/ACRS | 5200045 |
| 2/4/2019 | ML18164A109 | Letter to Applicant - GEH ABWR DC Renewal SE for Subsection 6.2.1. | Letter | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 2/7/2019 | ML18324A747 | GEH ABWR DC Renewal SE for Subsections 8.2.5 and 8.3.3. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DLSE | NRC/ACRS | 5200045 |
| 2/8/2019 | ML18170A118 | GEH ABWR DC Renewal Safety Evaluation for Subsection 6.2.1.6. | Safety Evaluation Report | NRC/NRO | NRC/ACRS | 5200045 |
| 2/8/2019 | ML18052A925 | ABWR SER Subsection 6.2.1.3 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DNRL | NRC/ACRS | 5200045 |
| 4/24/2019 | ML19091A121 | Letter to applicant - GEH ABWR DC Renewal Issue 22 SE for Section 7.7.1.2.1 CR Ganged Withdrawal Sequence | Letter | NRC/NRO/ DLSE | NRC/ACRS | 5200045 |
| 4/24/2019 | ML19091A122 | Memo - GEH ABWR DC Renewal Issue 22 SE for Section 7.7.1.2.1 CR Ganged Withdrawal Sequence. | Memoranda | | NRC/ACRS | 5200045 |
| 5/21/2019 | ML19091A120 | ABWR DC Renewal Issue 22 SE Section 7.7.1.2.1 CR Ganged Withdrawal Sequence. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ DLSE | NRC/ACRS | 5200045 |

| Document Date | Accession Number | Title | Document Type | Author Affiliation | Addressee Affiliation | Docket Number |
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| 5/29/2019 | ML19136A281 | Regulatory Audit Results Summary Report for the Advanced Boiling-Water Reactor Design Certification Renewal Peak Cladding Temperature Increase. | Audit Report | NRC/NRO | GEH | 5200045 |
| 5/31/2019 | ML19142A186 | GE Hitachi Nuclear Energy U. S. Advanced Boiling Water Reactor Design Certification Renewal Schedule Letter, May 2019 | Memoranda | NRC/NRO/ DLSE | GEH | 5200045 |
| 6/5/2019 | ML19148A514 | Memo - GEH ABWR DC Renewal Issue 26 SER for Section 5.4.7 Residual Heat Removal System. | Graphics incl Charts and Tables | NRC/NRO/ DLSE | | 5200045 |
| 6/5/2019 | ML19148A594 | Memo - GEH ABWR DC Renewal Issue 26 SER for Section 5.4.7.1.1.10 ACIWA. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRO Safety Evaluation Report (SER)-Delayed | 5200045 |
| 6/5/2019 | ML19148A779 | Memo - GEH ABWR DC Renewal Issue 26 SER for Section 7.4.1.4.4 Shutdown Panel. | DLSE | NRC/ACRS | DLSE | 5200045 |
| 6/5/2019 | ML19149A314 | Memo - GEH ABWR DC Renewal Issue 26 SER for Section 8.3.4.4 Class 1E Bus Isolation. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRO Safety Evaluation Report (SER)-Delayed | 5200045 |

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| 6/5/2019 | ML19114A339 | Memo - GEH ABWR DC Renewal Issue 27 SE for Section 7.5.2.1 Post Accident Monitoring System Revision 0. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/5/2019 | ML19114A350 | Memo - GEH ABWR DC Renewal Issue 27 SE for Section 9.1.3 Fuel Pool Cooling and Cleanup System Revision 0 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/5/2019 | ML19148A515 | Letter to Applicant - GEH ABWR DC Renewal Issue 26 SER for Section 5.4.7 Residual Heat Removal System | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/5/2019 | ML19148A593 | Letter to Applicant - GEH ABWR DC Renewal Issue 26 SER for Section 5.4.7.1.1.10 ACIWA | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/5/2019 | ML19148A778 | Letter to Applicant - GEH ABWR DC Renewal Issue 26 SER for Section 7.4.1.4.4 Shutdown Panel | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/5/2019 | ML19149A315 | Letter to Applicant - GEH ABWR DC Renewal Issue 26 SER for Section 8.3.4.4 Class 1E Bus Isolation. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/5/2019 | ML19113A176 | Letter to applicant - GEH ABWR DC Renewal Issue 27 SE for Section 3.2.3 Safety Classifications Revision 0 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 6/5/2019 | ML19114A351 | Letter to applicant - GEH ABWR DC Renewal Issue 27 SE for Section 9.1.3 Fuel Pool Cooling and Cleanup System Revision 0 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/6/2019 | ML19141A371 | Letter to applicant - GEH ABWR DC Renewal Issue 26 SE for Section 22 Fukushima Design Related Changes | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/6/2019 | ML19141A372 | Memo - GEH ABWR DC Renewal Issue 26 SE for Section 22 Fukushima Design Related Changes. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/12/2019 | ML19148A460 | Memo - GEH ABWR DC Renewal Issue 26 SER for Section 16 Technical Specifications. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/12/2019 | ML19155A208 | Memo - GEH ABWR DC Renewal Issue 38 SER for Section 6.3 Emergency Core Cooling Systems | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/12/2019 | ML19148A461 | Letter to applicant - GEH ABWR DC Renewal Issue 26 SER for Section 16 Technical Specifications | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/12/2019 | ML19156A180 | Letter to Applicant - GEH ABWR DC Renewal Issue 19 & 20 for Section 9.1.2.1 New and Spent Fuel Storage | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 6/12/2019 | ML19156A151 | Letter to Applicant - GEH ABWR DC Renewal Issue 18a for Section 4.2 Rx LOCA and Seismic Load | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/12/2019 | ML19156A182 | ABWR DCD Renewal Issue 19 & 20 SER Section 9.1.2.1 New and Spent Fuel Storage Revision 6. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/12/2019 | ML19156A152 | GEH ABWR DC Renewal Issue 18a SER for Section 4.2 Rx LOCA & Seismic Load | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
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| 6/19/2019 | ML19114A353 | ABWR DC Renewal SER Section 9.1.3 Fuel Pool Cooling and Cleanup System Pointer SER Rev 4 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
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| 6/19/2019 | ML19113A173 | ABWR DC Renewal SER Section 3.2.3 Safety Classifications Pointer SER Rev 4 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/19/2019 | ML19114A365 | ABWR DC Renewal SER Section 7.5.2.1 Post Accident Monitoring System Pointer SER Rev 3 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/19/2019 | ML19148A516 | ABWR DC Renewal Issue 26 Section 5.4.7 RHR Rev 5 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/19/2019 | ML19148A780 | ABWR DC Renewal Issue 26 Section 7.4.1.4.4 Shutdown Panel Inst Rev 4 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/19/2019 | ML19149A317 | ABWR DC Renewal Issue 26 Section 8.3.4.4 RB Class 1E Bus Isloation Rev 4 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/19/2019 | ML19162A080 | Letter to applicant - GEH ABWR DC Renewal Issue 09 Section 6.2.1.9 ECCS Strainers. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/19/2019 | ML19164A036 | Memo - GEH ABWR DC Renewal Section 1 Introduction and General Discussion | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 6/20/2019 | ML19148A592 | ABWR DC Renewal Issue 26 SER for Section 5.4.7.1.1.10 ACIWA Rev 9 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/20/2019 | ML19156A153 | ABWR DCD Renewal Issue 18a SER Section 4.2 Rx LOCA and Seismic Load | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/25/2019 | ML19155A207 | ABWR DC Renewal Issue 38 SE Section 6.3 ECCS Rev 3 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/26/2019 | ML19164A037 | ABWR DC Renewal SE Chapter 1 Revision 3 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/26/2019 | ML19162A078 | ABWR DC Renewal Issue 09 SE Section 6.2.1.9 ECCS Strainers Rev 5 | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/27/2019 | ML19171A293 | ABWR DC Renewal Issue 18b Section 9.1.2.2 New and SFP Racks Seismic Structural Analysis Rev 4. | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |
| 6/27/2019 | ML19171A291 | Letter to applicant - GEH ABWR DC Renewal Issue 18b for Section 9.1.2.2 New and SFP Racks Seismic Structural Analysis | NRO Safety Evaluation Report (SER)-Delayed | NRC/NRO/ | NRC/ACRS | 5200045 |

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| 10/31/2019 | ML20076D969 | 25A5675AG ABWR DCD Rev 7 Tier 2 Chapter 5 NP Revbar (Security-Related Information). | Design Control Document (DCD) | GE Hitachi Nuclear Energy | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E274 | GE Hitachi Nuclear Energy - ABWR Standard Plant Design Certification Renewal Application Design Control Document Revision 7, Tier 1 and Tier 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E086 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 01 - INTRODUCTION AND GENERAL DESCRIPTION OF PLANT | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E327 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 01 - INTRODUCTION AND GENERAL DESCRIPTION OF PLANT | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E087 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 02 - SITE CHARACTERISTICS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E328 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 02 - SITE CHARACTERISTICS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E089 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 03 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E331 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 03 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E090 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 04 - REACTOR | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E332 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 04 - REACTOR | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20104C141 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - Chapter 05 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E091 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 05 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E334 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 05 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20093F525 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - Chapter 05 - Reactor Coolant System and Connected Systems | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E093 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 06 - ENGINEERED SAFETY FEATURES | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E335 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 06 - ENGINEERED SAFETY FEATURES | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E095 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 07 - INSTRUMENTATION AND CONTROL SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E336 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 07 - INSTRUMENTATION AND CONTROL SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E096 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 08 - ELECTRIC POWER | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E337 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 08 - ELECTRIC POWER | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E098 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 09 - AUXILIARY SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E338 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 09 - AUXILIARY SYSTEMS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E099 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 10 - STEAM AND POWER CONVERSION SYSTEM | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E340 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 10 - STEAM AND POWER CONVERSION SYSTEM | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E100 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 11 - RADIOACTIVE WASTE MANAGEMENT | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E342 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 11 - RADIOACTIVE WASTE MANAGEMENT | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E101 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 12 - RADIATION PROTECTION | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E343 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 12 - RADIATION PROTECTION | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E102 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 13 - CONDUCT OF OPERATIONS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E344 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 13 - CONDUCT OF OPERATIONS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E103 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 14 - INITIAL TEST PROGRAM | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E345 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 14 - INITIAL TEST PROGRAM | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E105 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 15 - ACCIDENT AND ANALYSIS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E348 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 15 - ACCIDENT AND ANALYSIS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E108 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 16 - TECHNICAL SPECIFICATIONS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E349 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 16 - TECHNICAL SPECIFICATIONS | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E110 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 16 - TECHNICAL SPECIFICATIONS BASES | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E351 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 16 - TECHNICAL SPECIFICATIONS BASES | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E112 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 17 - QUALITY ASSURANCE | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E352 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 17 - QUALITY ASSURANCE | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E114 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 18 - HUMAN FACTORS ENGINEERING | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E354 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 18 - HUMAN FACTORS ENGINEERING | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E117 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 19 - RESPONSE TO SEVERE ACCIDENT POLICY STATEMENT | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E357 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 19 - RESPONSE TO SEVERE | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E119 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 20 - QUESTION AND RESPONSE GUIDE | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E359 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 20 - QUESTION AND RESPONSE GUIDE | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E123 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E361 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 1 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E126 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E362 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 2 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E128 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 3 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E363 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E129 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E365 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 4 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |

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| 12/20/2019 | ML20007E130 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 5 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
| 12/20/2019 | ML20007E366 | GE-Hitachi ABWR Design Control Document Tier 1 & 2, Rev. 7 - CHAPTER 21 - ENGINEERING DRAWINGS - VOLUME 5 | Design Control Document (DCD) | GE-Hitachi Nuclear Energy Americas, LLC | NRC/ Document Control Desk | 5200045 |
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# APPENDIX C
# ELECTRONIC REQUEST FOR ADDITIONAL INFORMATION

The following notes pertain to the table on the proceeding pages:

- The request for additional information (RAI) question numbers were assigned based on the section of the standard review plan (SRP) that was associated with the question (e.g., question 06.03-1 was generated based on the staff's review of the application against Section 6.3 of the SRP).
- The NRC letter number is a unique number that was assigned to the letter that transmitted the RAIs to the applicant.
- The applicant's responses to security-related and sensitive information questions (e.g., physical security) are not publicly available.

| Question Number | NRC Letter No. | System RAI No. | FSER Chapter | RAI Accession Number | RAI Response Date | Response Accession Number |
|---|---|---|---|---|---|---|
| 01.05-1 | 2 | 7653 | 2,12,22 | ML14267A352 | 11/6/2014 | ML14310A567 |
| 01.05-1 | 2 | 7653 | 2,12,22 | ML14267A352 | 6/18/2015 | ML15170A044 |
| 01.05-1 | 2 | 7653 | 2,12,22 | ML14267A352 | 8/25/2015 | ML15237A192 |
| 01.05-2 | EA-1 | 8721 | Other | ML17032A537 | 3/21/2017 | ML17080A065 |
| 01.05-3 | EA-1 | 8721 | Other | ML17032A537 | 3/21/2017 | ML17080A065 |
| 01.05-4 | EA-1 | 8721 | Other | ML17032A537 | 3/21/2017 | ML17080A065 |
| 02-1 | 2 | 7668 | 2 | ML14267A352 | 1/12/2016 | ML16012A290 |
| 02-1 | 2 | 7668 | 2 | ML14267A352 | 11/5/2015 | ML15309A157 |
| 02-1 | 2 | 7668 | 2 | ML14267A352 | 6/26/2015 | ML15177A036 |
| 02-1 | 2 | 7668 | 2 | ML14267A352 | 11/19/2014 | ML14324A082 |
| 02-1 | 2 | 7668 | 2 | ML14267A352 | 4/13/2017 | ML17103A123 |
| 02.05.04-1 | 8 | 7788 | 2 | ML15160A421 | 7/24/2015 | ML15209A561 |
| 02.05.04-1 | 8 | 7788 | 2 | ML15160A421 | 5/31/2016 | ML16152A512 |
| 02.05.04-1 | 8 | 7788 | 2 | ML15160A421 | 11/13/2015 | ML15317A092 |
| 06.02.01.01. C-1 | 1 | 7230 | 6 | ML14114A566 | 8/27/2014 | ML14239A137 |
| 06.02.01.01. C-1 | 1 | 7230 | 6 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 06.02.01.01. C-1 | 1 | 7230 | 6 | ML14114A566 | 8/11/2015 | ML15223B146 |
| 06.02.01.01. C-1 | 1 | 7230 | 6 | ML14114A566 | 6/22/2016 | ML16174A179 |
| 06.02.01-1 | 5 | 7797 | 6 | ML15110A122 | 5/29/2015 | ML15149A232 |
| 06.02.02-1 | 12 | 8834 | 6 | ML17130A798 | 6/16/2017 | ML17167A161 |
| 06.03-1 | 3 | 7795 | 6 | ML15068A227 | 7/17/2015 | ML15198A332 |
| 06.03-1 | 3 | 7795 | 6 | ML15068A227 | 4/8/2015 | ML15098A484 |
| 06.03-2 | 9 | 8288 | 6 | ML15343A408 | 12/19/2016 | ML16358A445 |
| 06.03-2 | 9 | 8288 | 6 | ML15343A408 | 5/27/2016 | ML16148A101 |
| 06.03-3 | 11 | 8733 | 6 | ML17087A290 | 8/23/2017 | ML17236A059 |
| 06.03-3 | 11 | 8733 | 6 | ML17087A290 | 4/25/2017 | ML17116A071 |

| Question Number | NRC Letter No. | System RAI No. | FSER Chapter | RAI Accession Number | RAI Response Date | Response Accession Number |
|---|---|---|---|---|---|---|
| 06.03-4 | 13 | 8799 | 6 | ML17187A127 | 8/23/2017 | ML17236A059 |
| 06.03-5 | 13 | 8799 | 6 | ML17187A127 | 8/23/2017 | ML17236A059 |
| 06.03-6 | 13 | 8799 | 6 | ML17187A127 | 8/23/2017 | ML17236A059 |
| 06.03-7 | 13 | 8799 | 6 | ML17187A127 | 8/23/2017 | ML17236A059 |
| 06.03-8 | 13 | 8799 | 6 | ML17187A127 | 8/23/2017 | ML17236A059 |
| 06.03-9 | 13 | 8799 | 6 | ML17187A127 | 8/23/2017 | ML17236A059 |
| 07-1 | 5 | 7658 | 7 | ML15110A122 | 5/19/2015 | ML15139A210 |
| 08.02-1 | 1 | 7435 | 8 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 08.02-1 | 1 | 7435 | 8 | ML14114A566 | 8/27/2014 | ML14241A556 |
| 08.02-2 | 7 | 7865 | 8 | ML15154B692 | 7/14/2015 | ML011234567 |
| 08.02-2 | 7 | 7865 | 8 | ML15154B692 | 9/25/2015 | ML15271A170 |
| 09.05.01-1 | 6 | 7665 | 9 | ML15118A725 | 7/30/2015 | ML15212A762 |
| 09.05.01-1 | 6 | 7665 | 9 | ML15118A725 | 10/29/2015 | ML15302A308 |
| 09.05.01-1 | 6 | 7665 | 9 | ML15118A725 | 4/11/2016 | ML16102A344 |
| 11.04-1 | 4 | 7781 | 11 | ML15069A674 | 4/9/2015 | ML15099A586 |
| 11.04-1 | 4 | 7781 | 11 | ML15069A674 | 7/21/2015 | ML15202A045 |
| 12.02-1 | 2 | 7583 | 12 | ML14267A352 | 12/16/2014 | ML14350A843 |
| 12.02-2 | 2 | 7583 | 12 | ML14267A352 | 1/22/2015 | ML15023A016 |
| 12.02-2 | 2 | 7583 | 12 | ML14267A352 | 7/7/2015 | ML15194A053 |
| 12.02-3 | 2 | 7583 | 12 | ML14267A352 | 1/22/2015 | ML15023A016 |
| 12.02-3 | 2 | 7583 | 12 | ML14267A352 | 7/7/2015 | ML15194A053 |
| 13.03-1 | 10 | 8606 | 13 | ML16160A067 | 6/28/2016 | ML16180A256 |
| 14.03-1 | 3 | 7787 | 14 | ML15068A227 | 4/1/2015 | ML15092A175 |
| 19-1 | 1 | 7125 | 19 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 19-1 | 1 | 7125 | 19 | ML14114A566 | 9/24/2014 | ML14273A455 |
| 19-2 | 1 | 7125 | 19 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 19-2 | 1 | 7125 | 19 | ML14114A566 | 9/24/2014 | ML14273A455 |
| 19-3 | 1 | 7125 | 19 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 19-3 | 1 | 7125 | 19 | ML14114A566 | 9/24/2014 | ML14273A455 |
| 19-4 | 1 | 7125 | 19 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 19-4 | 1 | 7125 | 19 | ML14114A566 | 9/24/2014 | ML14273A455 |
| 19-5 | 1 | 7125 | 19 | ML14114A566 | 7/16/2014 | ML14197A127 |
| 19-5 | 1 | 7125 | 19 | ML14114A566 | 9/24/2014 | ML14273A455 |
| 19-5 | 1 | 7125 | 19 | ML14114A566 | 11/3/2014 | ML14309A023 |
| 19-5 | 1 | 7125 | 19 | ML14114A566 | 7/7/2015 | ML15188A260 |
| 19-5 | 1 | 7125 | 19 | ML14114A566 | 1/22/2016 | ML16022A252 |
| 19-6 | 5 | 7826 | 19 | ML15110A122 | 7/17/2015 | ML15198A101 |
| 19-6 | 5 | 7826 | 19 | ML15110A122 | 9/17/2015 | ML15264A003 |
| 19-7 | 5 | 7826 | 19 | ML15110A122 | 7/17/2015 | ML15198A101 |
| 19-7 | 5 | 7826 | 19 | ML15110A122 | 9/17/2015 | ML15264A003 |

| Question Number | NRC Letter No. | System RAI No. | FSER Chapter | RAI Accession Number | RAI Response Date | Response Accession Number |
|---|---|---|---|---|---|---|
| 19-8 | 5 | 7826 | 19 | ML15110A122 | 7/17/2015 | ML15198A101 |
| 19-8 | 5 | 7826 | 19 | ML15110A122 | 9/17/2015 | ML15264A003 |
| 19-9 | 5 | 7826 | 19 | ML15110A122 | 7/17/2015 | ML15198A101 |
| 19-9 | 5 | 7826 | 19 | ML15110A122 | 9/17/2015 | ML15264A003 |
| 19-9 | 5 | 7826 | 19 | ML15110A122 | 11/19/2015 | ML15323A354 |

# APPENDIX D
# PRINCIPAL CONTRIBUTORS

| Name | Responsibility |
|---|---|
| | |
| Anderson, Joseph | Emergency Planning |
| Barss, Daniel | Emergency Planning |
| Basturescu, Sergiu | Instruments and Controls |
| Budzynski, John | Reactor Systems |
| Caruso, Mark | PRA/Severe Accidents |
| Cushing, John S | Environmental |
| Dias, Antonio | Plant Systems |
| Dudek, Michael I | Project Management |
| Ezell Julie | OGC |
| Giacinto, Joseph | Hydrology |
| Gilmer, James | Reactor Systems |
| Green, Sharon | Licensing Assistant |
| Hansing, Nicholas James | Mechanical Engineering |
| Harbuck, Craig | Technical Specifications, Regulatory Treatment of Non-Safety Systems, Availability Controls |
| Hayes, Michelle | PRA |
| Hart, Michelle | Accident Analysis |
| Harvey, Brad | Meteorology |
| Heida, Bruce | Plant Systems |
| Hernandez, Raul | Plant Systems |
| Hernandez,Raul | Plant Systems Mitigating Strategies |
| Honcharik, John | Environmental |
| Huang, Jason | Mechanical Engineering |
| Hsueh, Kevin | Radiation Control |
| Istar, Ata | Structural Engineering |
| Johnston, Jeanne | Instrument and Controls |
| Le, Hien | Plant Systems |
| Li, Chang | Plant Systems |
| Li, Yueh-Li | Mechanical Engineering |
| Li, Zuhan | Geotechnical Engineering |
| Li,Chang-yang | Plant Systems |
| Li,Huan | Structural Engineering |
| Li, Yueh-li C | Mechanical Engineering |
| Lupold, Timothy | Mechanical Engineering |
| Krepel, Scott | Technical Specifications |
| Makar, Gregory | Materials Engineering |
| Martin, Jody | OGC |
| Martinez-Navedo, Tania | Electrical Engineering |
| Mikula Olivia | OGC |

| Name | Responsibility |
|---|---|
| Miller, Joshua | Reactor Systems |
| Muniz Gonzalez, Adrian | Project Management |
| Musico, Bruce J | Emergency Planning |
| Nakanishi, Tony | PRA, Plant Procedures |
| Nolan, Ryan | Plant Systems |
| Otto, Ngola | Electrical Engineering |
| Patterson, Malcolm | PRA |
| Patton, Rebecca | Reactor Systems |
| Palmrose, Donald E. | Environmental Evaluation of NPSI |
| Patel, Raju B | Plant Systems |
| Phan, Hanh | PRA |
| Quinlan, Kevin R. | Meteorology |
| Rankine, Jennivine | Project Management |
| Ray, Sheila | Electrical Engineering |
| Reed, Addison | Reactor Systems |
| Roach, Kevin | OGC |
| Sacko, Fanta | Electrical Engineering |
| Salgado, Nancy | Instruments and Controls |
| Scarbrough, Thomas | Mechanical Engineering |
| Shea, James | Project Management |
| Shukla, Girija S | ACRS Project Manager |
| Spencer, Michael | OGC |
| Stutzcage, Edward | Radiation Control |
| Tammara, Seshagiri Rao | Site Hazards |
| Taneja, Dinesh | Instrumentation and Control |
| Tesfaye, Getachew | Project Management |
| Thomas, Mathew | Reactor Systems |
| Tjader,Theodore R | Plant Systems Tech Specs |
| Tseng, Ian | Mechanical Engineering |
| Vettori, Robert L | Plant Systems Fire Protection |
| Whitman, Jennifer | Plant Systems Fire Progectoin |
| Wagage, Hanry | Containment Analysis |
| Wang, Weijun | Geotechnical Engineering |
| White, Jason D. | Meteorology |
| Williams, Donna | Structural Engineering |
| Wilson, Anthony | OGC |
| Wittick, Brian | Mechanical Engineering |
| Wong, Yuken | Mechanical Engineering |
| Wu, Cheng-Ih | Mechanical Engineering |
| Xi, Zuhan | Reactor Systems |
| Zhao, Jack Y | Digital I&C |

# APPENDIX E
# ACRS LETTER

**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**
**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**WASHINGTON, DC 20555 - 0001**

October 31, 2019

The Honorable Kristine L. Svinicki
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: ADVANCED BOILING WATER REACTOR DESIGN CERTIFICATION RENEWAL

Dear Chairman Svinicki:

During the 667th meeting of the Advisory Committee on Reactor Safeguards (ACRS), October 2-4, 2019, we completed our review of the design certification renewal application for the advanced boiling water reactor (ABWR) and the associated final safety evaluation report. Our review considered actions by GE-Hitachi (GEH), the first vendor in the U.S. to apply for a design certification renewal. Our ABWR Subcommittee reviewed this matter during a meeting on August 23, 2019. During our review, we had the benefit of discussions with representatives of the staff and GEH. We also had the benefit of the referenced documents.

This report fulfills the requirement of Title 10 of the *Code of Federal Regulations* (10 CFR) 52.57(c) that the ACRS report on those portions of the application which concern safety.

**CONCLUSION AND RECOMMENDATION**

Staff supplemental safety evaluations (SEs) approved GEH proposed design changes to update and amend specific design attributes that meet the criteria for a Design Certification Renewal in accordance with 10 CFR § 52.59, extending it for an additional 15 years, following implementation of the design certification final rule.

1. There is reasonable assurance that the ABWR, under the renewed design certification, can be constructed and operated without undue risk to the health and safety of the public.

2. We concur with the conclusions of the staffs' supplemental renewal SEs to NUREG–1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," with no open items. The SEs should be issued, and the GEH application for the Design Certification Renewal of the ABWR should be approved.

## BACKGROUND

Previously, on July 13, 1994, the U.S. Nuclear Regulatory Commission (NRC) issued the final design approval, along with NUREG–1503, “Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design.” On May 12, 1997, the NRC issued the final design certification rule for the ABWR design.

On December 7, 2010, GEH requested the NRC to renew the ABWR design certification. The ABWR design certification rule, effective June 11, 1997, would otherwise expire at the end of a period of 15 years, or June 11, 2012. GEH applied for a design certification renewal on December 7, 2010. On July 20, 2012, staff identified proposed changes including Fukushima Near Term Task Force Recommendations. GEH provided the ABWR design control document (DCD), Revision 6, in response to staff requested changes. On June 28, 2019, the staff completed the SEs with no open items.

## DISCUSSION

The regulatory basis for renewal of a design certification includes three change categories: modifications, renewal backfits, and amendments. Modifications to the certified design are those changes in accordance with 10 CFR § 52.57(a) (e.g., clarifications, changes to correct known errors, typographical errors, or defects that are necessary to meet 10 CFR § 52.59(a)). Modifications must comply with the regulations applicable and in effect at the time the certification was originally issued. Renewal backfits are those changes that are necessary to comply with additional requirements imposed by the NRC through application of the criteria in 10 CFR § 52.59(b). Amendments are those changes proposed by the design certification renewal applicant in accordance with 10 CFR § 52.59(c). Amendments must comply with regulations applicable and in effect at the time of renewal. The GEH Design Certification renewal application contains modifications and amendments but no backfits.

The key significant renewal design changes involved the following areas: amendment to the emergency core cooling system (ECCS) suction strainers; peak cladding temperature (PCT) modification; Fukushima design enhancements; aircraft impact assessment; and containment overpressure protection system (COPS) modification.

- In accordance with guidance of Regulatory Guide 1.82, Revision 4, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” the staff confirmed that the ECCS suction strainer design complied with 10 CFR § 50.46(b)(5), providing adequate Net Positive Suction Head margins. The staff also confirmed that GEH addressed the chemical, in-vessel, and ex-vessel downstream effects.
- Following incorporation of the effects of the ECCS evaluation model changes, and correction of errors since the original ABWR design certification, the estimated PCT increased by a small amount (42°C or 75°F). PCT is now 663 °C (1225 °F), which remains in compliance with criteria in 10 CFR § 50.46(3)(i).
- To allow combined license applicants to meet anticipated requirements of the Mitigation of Beyond-Design-Basis Events Rule, GEH made design amendments, such as additional non-safety-related water and electrical connections.
- GEH performed a detailed aircraft impact assessment. The staff found that GEH adequately described the key design features and functional capabilities identified and credited to meet 10 CFR § 50.150(b), including how the key design features meet the acceptance criteria in 10 CFR § 50.150(a)(1).

- GEH modified the COPS design to include a dedicated containment vent path to prevent containment over pressure. The staff concluded that this modification did not alter the safety findings made in NUREG-1503.

In total, 39 design items were reviewed and approved by the staff in supplemental SEs to NUREG-1503 or closed by letter. In addition to reviewing DCD, Revision 6, and responses to requests for additional information, the staff performed audits to resolve outstanding technical issues.

**SUMMARY**

The staff made safety determinations on the specific modifications and amendments proposed by GEH as part of its design certification renewal application; they were found to meet applicable regulatory requirements. We agree with the staff's determinations. There is reasonable assurance that the ABWR, under the renewed design certification, can be constructed and operated without undue risk to the health and safety of the public.

We are not requesting a formal response from the staff to this letter report.

Sincerely,

**/RA/**

Peter C. Riccardella
Chairman

**REFERENCES**


1. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, Main Report," July 1994 (ML080670592).

2. W. T. Russell (NRC), letter providing final design approval (FDA) for the ABWR design to GE, July 13, 1994, fiche: 80268: 037-043.

3. U.S. Nuclear Regulatory Commission, "10 CFR Part 52, Appendix A, 'Design Certification Rule for the U.S. Advanced Boiling-Water Reactor'," May 12, 1997.

4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 02 SER Section 2.3 Meteorology," June 20, 2018 (ML18026A750).

5. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 01 SER Section 2.5 Geological, Seismological, and Geotechnical Engineering," June 12, 2017 (ML17060A378).

6. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 04 SER Section 2.6.2 Water Level (Flood) Design Site Parameters," December 11, 2017 (ML17080A134).

7. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 03 SER Section 2.6.8 Requirements for Determination of ABWR Site Acceptability," April 11, 2017 (ML17065A316).

8. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 27 Enhancements related to Fukushima Recommendation 7.1 Spent Fuel Pool Instruments SER Section 3.2.3 Safety Classifications," June 19, 2019 (ML19113A173).

9. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 02 SER Section 3.3 Wind and Tornado Loadings," June 20, 2018 (ML18026A667).

10. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 02 SER Section 3.5.1.4 Missiles Generated by Natural Phenomena," June 20, 2018 (ML18026A776).

11. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 23 SER Section 3.7.3 Seismic Subsystem Analysis," July 26, 2018 (ML18029A130).

12. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 18a SER Section 4.2 Fuel System Design,” June 20, 2019 (ML19156A153).

13. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 12 SER Section 5.2.5 Reactor Coolant Pressure Boundary Leakage Detection," May 7, 2018 (ML18052A137).

14. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 26 Fukushima Related Design Enhancements SER Section 5.4.7 Residual Heat Removal System," June 19, 2019 (ML19148A516).

15. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 26 Fukushima Related Design Enhancements SER Section 5.4.7.1.1 Alternating Current Independent Water Addition," June 20, 2019 (ML19148A592).

16. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 10 SER Section 5.4.8 Reactor Water Cleanup System," January 15, 2019 (ML18346A609).

17. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 30 SER Section 6.2.1.3 Short-Term Pressure Response," February 8, 2019 (ML18052A925).

18. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 34 SER Section 6.2.1.6 Suppression Pool Dynamic Loads," February 8, 2019 (ML18170A118).

19. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 09 SER Section 6.2.1.9 Containment Debris Protection for ECCS Strainers," June 26, 2019 (ML19162A078).

20. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items SER Section 6.3 Emergency Core Cooling Systems,” June 25, 2019 (ML19155A207).

21. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 26 Fukushima Related Design Enhancements SER Section 7.4.1.4.4 Shutdown Panel," June 19, 2019 (ML19148A780).

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23. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 22 SER Section 7.7.1.2.1 Control Rod Ganged Withdrawal Sequence Restrictions," May 21, 2019 (ML19091A120).

24. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 33 SER Section 8.2.5 NRC Bulletin 2012-01: Design Vulnerability in Electric Power System,” February 7, 2019 (ML18324A747).

25. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 26 Fukushima Related Design Enhancements SER Section 8.3.4.4 Isolation Between Class 1E Buses and Loads Designated as Non-Class 1E," June 19, 2019 (ML19149A317).

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27. U.S. Nuclear Regulatory Commission, “Safety Evaluation Report with No Open Items Related to the ABWR DC Renewal Issue 19 & 20 SER Section 9.1.2.1 New and Spent Fuel Storage," June 12, 2019 (ML19156A182).

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| NRC FORM 335 (12-2010) NRCMD 3.7 **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)* | U.S. NUCLEAR REGULATORY COMMISSION | 1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1503, Supplement 2 | |
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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Appendix A, “Design Certification Rule for the U.S. Advanced Boiling Water Reactor,” to Title 10 of the Code of Federal Regulations (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” constitutes the standard design certification (DC) for the U.S. Advanced Boiling-Water Reactor (ABWR) design. To document the U.S. Nuclear Regulatory Commission (NRC) staff’s review supporting initial certification of the ABWR, the staff issued a final safety evaluation report (FSER) in NUREG-1503, “Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design,” in July 1994 and NUREG-1503, Supplement 1, in May 1997.

In a letter dated December 7, 2010 (Agencywide Documents Access and Management System Accession No. ML110040176), GE-Hitachi Nuclear Energy (GEH or the applicant) submitted a Design Certification (DC) renewal application for the ABWR pursuant to the requirements of Subpart B, “Standard Design Certifications,” 10 CFR Part 52.

This supplemental FSER (Supplement 2 to NUREG-1503) documents the NRC or the Commission staff’s technical review.

| 12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.) | 13. AVAILABILITY STATEMENT |
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